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July, 1965

MASTER

Westinghouse Astronuclear Laboratory

THE COUPLED NOFLOW-FIPDIF COMPUTER PROGRAM

(Title Unclassified)

TME 1224 July 65

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THE COUPLED NOFLOW-FIPDIF COMPUTER PROGRAM

(Title Unclassified)

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I. SUMMARY

This report describes the NOFLOW-FIPDIF computer program which has been written to perform the following two related functions:

(1) Determination of the temperatures attained by reactor materials following a loss of coolant accident.

(2) Evaluation of the fission product inventory which would exist at any time (following loss of coolant).

In order that these objectives may be accomplished, heat transfer calculations are performed for each of sixteen sections of the core, tie rods, filler strips, reflectors, and the pressure vessel. Other components are divided into an appropriate number of sections. The program has been designed to calculate the fission product inventory due to eighty decay chains in each core section. Thus, the effects of axial power and temperature variation may be reflected in the decay heat source calculations.

The program has been used to simulate accidents involving loss of coolant during the initial firing of a nuclear reactor and during a re-start run. Four periods of power operation were considered in the cases involving accidents during initial firings. These ranged from 63 seconds to 613 seconds. Four similar periods of re-start operation were considered. It was assumed that the re-start commenced after the reactor was shutdown for a period of 5288 seconds following a 562 second operation at full power.

II. INTRODUCTION

This report describes a digital computer chain program that has been produced by linking modified versions of two other computer programs, NOFLOW[1] and FIPDIF[2].

The original NOFLOW program was designed to determine the effect, on reactor components, of an accident which resulted in a cessation of hydrogen coolant flow to the reactor after a prescribed period of full-power operation. The NOFLOW program performs calculations of reactor component temperatures changes by doing a heat balance involving net conduction and radiation of heat to (or from) the component section, heat deposition in the section due to fission power, and deposition due to decay of fission products. Calculation of the decay heat source requires that the fission product inventory be known. This inventory varies with position in the core because of significant axial power and temperature variations. The original NOFLOW program represents the decay heat source as the sum of points obtained from curves of decay heat generation vs time after shutdown for various periods of reactor operation. It does not attempt to account for the selective diffusion of the isotopes which comprise the heat source.

FIPDIF is a program which was written to provide information concerning the inventory of fission products released to the atmosphere by diffusion and that retained in the fuel for any prescribed set of reactor operating conditions. (Providing that fuel temperature remains less than that required for sublimation.)

Since the temperature profile which would occur following a loss of coolant accident is a function of the fission product inventory and, similarly, the fission product inventory at any time in any given core location is a function of the temperature history at that position, it was necessary to combine FIPDIF and NOFLOW in order to obtain meaningful temperatures and fission product data when simulating a loss of coolant accident. Because each of these programs require more than two thirds of the available computer storage, it was necessary to divide their computational functions between two links of a chain program and to separate some of the input and initializing operations performed by both programs into a third link.

The sequence of operations performed by the linked program is as follows:

1. LINK I causes all necessary input to be read, sets values for required constants, and provides initial values for variables used in LINKS II and III. Control is then transferred to LINK III.
2. LINK III (FIPDIF) calculates the fission product inventories retained and released during the period of normal operation and an approximate rate at which the decay power is diminishing at the time that loss of coolant occurs. Control is then transferred to LINK II.

3. LINK II (NOFLOW) calculates temperature changes in the reactor components for a series of time steps following transfer from LINK III. The decay power used in the heat balance calculations during this time interval is extrapolated from the last value of the decay power and the rate at which it diminishes, as determined in LINK III. Calculations, performed by LINK II, proceed until the temperature change in any core section differs by 400°R from that which existed at the time transfer was made from LINK III. Time-weighted core section temperatures and power levels are then computed for this time interval (ΔT2) and control is again transferred to LINK III.

4. LINK III calculates changes in fission product inventories retained and released to the atmosphere during the time interval ΔT2. A decay power is calculated which corresponds to the fission product inventory retained. Transfer is returned to LINK II.

5. LINK II calculates temperature changes for a single time step (DELT) and transfer is returned to LINK III.

6. In order to determine the rate at which the decay power diminishes, LINK III calculates another value of the decay power corresponding to the retained inventories at the

midpoint of DELT. Transfer is then returned to LINK II.

Steps (3) through (6) are repeated until the accident simulation is completed. At this time, transfer is made to LINK I which causes the power and temperature history of the reactor components to be written on the output tape.

This report is divided into two parts and three (3) appendices. The first part contains a discussion of the operations performed by each link. The second part gives results of a simulation of an accident involving loss of coolant following a reactor operation of 613 seconds duration, using this program. Appendix A contains a compilation of flow diagrams illustrating the sequence of operations performed by the routines which comprise each link. Appendix B is a list of the input required to simulate a loss of coolant accident. Appendix C contains a description of NFPLOT, a program which has been written to permit SC-4020 curve plotting from the data tape prepared by LINK II.

NFPLOT will prepare either of the two following types of curves:

1. At any desired output time, a series of curves of the axial temperature profile in each reactor component, and
2. At any axial location in a given reactor component, a curve of temperature vs time following loss of coolant.

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1.0 MATHEMATICAL MODELS AND PROGRAM CONFIGURATION

1.1 LINK I

LINK I has been written to provide the input and initialization functions required by LINK II and LINK III. In addition, it controls the printing of data, calculated in LINK II, on the output tape. MAIN I is used only twice during the simulation of a loss of coolant accident and the subroutines of this link are used only once. The functions performed by LINK I are therefore easily separable from the calculationsl scheme and the separation has resulted in considerable savings in computer storage and running time.

A brief discussion of the operations performed by the main program and the subroutines which comprise this link is given below. Flow diagrams which illustrate the sequence of operations are given in Appendix A.

1.1.1 MAIN I

MAIN I is the controlling program for the first link of the chain program. If INDEX = 0, MAIN I reads data and sets values of constants necessary for simulation of a loss of coolant accident. This routine is not called again until the simulation has been completed (INDEX = 5). At this time it calls the OUTPUT subroutine. This routine causes temperature and power data, which are written on an intermediate tape during the run, to be printed on the output tape.

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1.1.2 FIPIN

FIPIN is used to read data necessary for fission product inventory and diffusion calculations, including power and temperature profiles for the reactor run time prior to loss of coolant. It is possible to specify the use of either an existing library tape or that the nuclide data be read from cards. If initial isotope concentrations are not input, the concentration of each isotope is set equal to zero in each core section.

1.1.3 DOMNOZ

DOMNOZ provides standard constants for use in calculations performed by the DOME and NOZZLE subroutines. This routine is called if the indicator NDATA4 is set equal to zero. If NDATA4 is set equal to 1, constants for DOME and NOZZLE calculations are read from cards in MAIN I.

1.1.4 NTEMP(M)

NTEMP(M) provides standard temperatures and power distributions corresponding to NRX-A full-power operating conditions. When the indicator NDATA1 is set equal to zero, NTEMP(1) is called. This defines the standard temperature distribution. When the indicator NDATA2 is set equal to zero, NTEMP(2) is called. This defines the standard distribution for deposition of power in each of the reactor components. If either of these indicators are set equal to one, the corresponding data are read from cards in

MAIN I.

1.1.5 OUTPUT

OUTPUT is used to print temperature distribution and power data that was written on the plotting tape during execution of the program. This routine has been separated from LINK II so that computer storage could be conserved in LINK II.

1.2 LINK II

The following operations are performed by subroutines which are part of LINK II:

(1) Heat transfer and temperature calculations for the core and reactor components vs time after loss of coolant,

(2) Calculation of fission power vs time after loss of coolant,

(3) Extrapolation of fission product decay power from the last two calculations performed by LINK III.

This part of the program is derived from and retains many of the features of the Thermal and Nuclear Transients (TNT) Program developed by W. Knecht and W. L. Howarth[3,4].

1.2.1 THE GEOMETRIC MODEL AND MODES OF HEAT TRANSFER CONSIDERED

The geometric model of the reactor which is used as the basis for all calculations is described in Reference (3). "Typical" flow channels have been selected in each component and the reactor described in terms of these channels. Figure 1.1 is an illustration of the relative location of components referred to in this discussion.

1.2.1.1 NOZZLE CHANNEL MODEL

The nozzle channel is divided into 21 segments of varying lengths. The distance of the point of intersection of each segment from the nozzle tip, thickness of aluminum backing in each region, surface area of the channel exposed to the core face for each segment, and the ratio of the area of a cross-section of the nozzle at each location to the cross-sectional area of the throat are required geometrical constants. The radiation shape factor and emissivity for each segment are also required. The most recent nozzle geometry data (AGC Dwg. No. 707629) has been incorporated in the program.

The modes of heat transfer considered in computing the temperature of each segment are:

1. Conduction of heat to (or from) each adjacent segment,
2. Radiation of heat from the tube to the backing of the nozzle,
3. Radiative heat transfer from the core bottom to the nozzle, and
4. Radiative heat loss from the outside of the nozzle to

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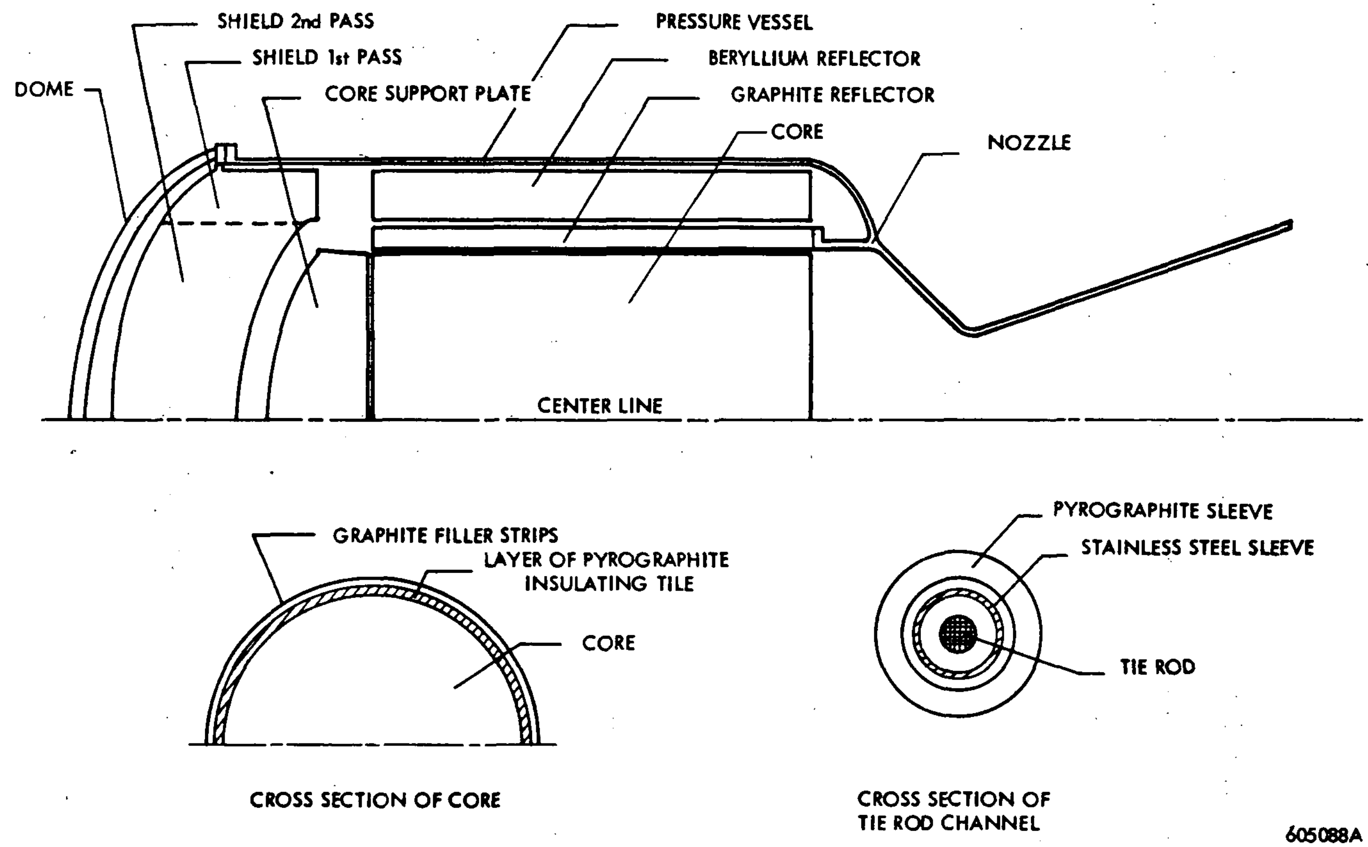

FIGURE 1.1

GEOMETRIC MODEL OF NERVA REACTOR

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the environment.

1.2.1.2 THE PRESSURE VESSEL MODEL

Since the pressure vessel contains no coolant channels, it is considered as a unit. The required geometric and physical property constants are the inner and outer radii of the vessel, the density of the material, the emissivity of the outer surface, and the emissivity and shape factor constants for the annulus between the pressure vessel and the beryllium reflector. The model considers the pressure vessel to be made up of an arbitrary number of segments (this number must be the same as the number chosen for the core channel). The modes of heat transfer considered in computing the temperature of any segment are:

1. Nuclear energy deposited in the segment,
2. Axial conduction of heat from (or to) adjacent segments,
3. Radiative heat transfer between the beryllium reflector and the pressure vessel, and
4. Radiative heat transfer to the environment.

Provision has been included for carrying out the calculation for either aluminum or titanium as the pressure vessel material, at the option of the program user.

1.2.1.3 BERYLLIUM REFLECTOR CHANNEL MODEL

Each channel in the beryllium reflector has been assumed to be identical, i.e., no distinction between channels in the reflector and channels in the control drum has been made. The true coolant channel radius has been used and an "effective" outer radius of the channel has been calculated such that when the weight of beryllium associated with each channel is multiplied by the actual number of channels the correct total reflector (with control drums) weight is obtained. The input constants required are inner and effective outer radius of the channel, inner and outer radius of the reflector measured from the center of the core, density of beryllium, and emissivity and shape factor constants for the annulus between graphite and beryllium reflectors. The channel is segmented in the same fashion as the pressure vessel. The modes of heat transfer considered in computing the temperature of each segment are:

1. Nuclear energy deposited in the segment. (The energy released due to the (n,α) reaction occurring in the boronated aluminum control vane is a part of the energy source for the reflector segment in the present model.),
2. Axial conduction of heat from (or to) adjacent segments,
3. Radiative heat transfer between graphite and beryllium reflectors, and
4. Radiative heat transfer between beryllium reflector and pressure vessel.

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1.2.1.4 GRAPHITE REFLECTOR CHANNEL MODEL

As in the case of the beryllium reflector, the true coolant channel radius has been used as the inner radius of the channel model and "effective" outer radius calculated such that when the channel weight is multiplied by the actual number of channels, the total reflector weight is obtained. The required constants are inner and effective outer radius of the channel, inner and outer radius of the reflector measured from the center of the core, emissivity and shape factor constants for the annulus between the graphite filler strips surrounding the core and the graphite reflector, and the density of graphite. The modes of heat transfer considered in computing the temperature of each segment of the graphite reflector channel are:

1. Nuclear energy deposited in the segment,
2. Axial conduction from (or to) adjacent segments,
3. Radiative heat transfer between the graphite filler strips and the graphite reflector, and
4. Radiative heat transfer between the graphite and beryllium reflectors.

1.2.1.5 CORE AND CORE CHANNEL MODEL

The core is considered to be surrounded by a layer of pyrographite tile which in turn is surrounded by the layer of unfueled graphite filler strips as shown in Figure 1.1. For simplicity, it has been

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assumed that these layers are cylindrical and concentric.

The same scheme has been used as in the reflectors to calculate an "effective" outer radius for the core coolant channel model, i.e., the true coolant channel inner radius is taken and an outer radius computed such that when the channel weight is calculated from these radii, the density of fueled graphite, and the channel length and then multiplied by the actual number of coolant channels, the true core weight is obtained. The constants required for the core calculations are the core radius, the outer radius of the layer of pyrographite tile, the outer radius of the graphite filler strip layer, the density of fueled graphite, the inner and effective outer radii of the coolant channel and the core length (the lengths of the graphite and beryllium reflectors are also taken to have this value). The core is cut into an arbitrary number of segments, and the segment length is computed from the chosen number and the known core length. This number, then, is the segment length for all the outer components also. In the present model, the total core length includes that of the unfueled core support blocks which face the nozzle.

The modes of heat transfer considered in computing the temperature of each segment of the core channel are:

1. Heat loss by conduction through the layer of pyrographite tile (this loss is computed for the whole core segment and then averaged over the total number of coolant

channels to obtain the loss for a single typical channel.),

2. Heat loss by conduction to the tie rod channel (This calculation is actually carried out along with all the calculations for tie-rod temperatures.),
3. Radiative heat transfer between the core inlet segment and the core support plate,
4. Radiative heat transfer between the core exit segment and the nozzle,
5. Axial conduction of heat from (and to) adjacent core channel segments,
6. Heat loss associated with sublimation of fuel, and
7. Nuclear energy deposited in the segment.

The temperature of each graphite filler strip segment is also computed and the modes of heat transfer considered in the calculation are:

1. Conduction of heat from the core,
2. Radiative heat transfer between the filler strips and the graphite reflector and,
3. Axial conduction of heat from (or to) adjacent filler strip segments.

1.2.1.6 TIE-ROD CHANNEL MODEL

The tie-rod channel model presumes a layer of pyrographite insulation between its inner components and the core, a stainless steel sleeve which is <u>not</u> in contact with the pyrographite, and the tie-rod itself. The orientation of this model is shown in Figure 1.1. All three components of the channel are assumed to be concentric cylinders. The outer surface of the pyrographite cylinder is assumed to be in contact with fuel material and to be at the same temperature as the fuel. The temperature at the middle of the pyrographite layer is computed as is the temperature of the inner face of the cylinder. The constants required for the calculations are the tie-rod radius, the inner and outer radii of the pyrographite cylinder, the inner and outer radii of the steel sleeve, the densities of pyrographite, stainless steel, and Inconel-X, and the emissivity and shape factors for the annulus between the pyrographite and steel cylinders as well as those for the annulus between the steel cylinder and the tie-rod.

The modes of heat transfer considered in computing the temperature of the components of the tie-rod model for each segment are:

1. Conduction of heat through the layer of pyrographite from the core,
2. Radiative heat transfer between the inner surface of the pyrographite and the steel sleeve,
3. Radiative heat transfer between the steel sleeve and tie-rod,
4. Deposition of nuclear energy in the tie-rod,

5. Axial conduction of heat from (or to) adjacent tie-rod segments (axial conduction is not considered for the steel sleeve or pyrographite cylinders), and
6. Heat required for melting at appropriate temperatures.

1.2.1.7 THE SHIELD AND SUPPORT PLATE MODEL

The shield and support plate channel is divided into three distinct parts. The first section of the channel is used to compute temperature profiles for the portion of the aluminum shield in which, if there were propellant flowing, the flow would be toward the dome, called the shield-first pass. The second section generates temperature profiles for the shield in which flow would be away from the dome and toward the core, termed the shield-second pass. The third section generates the time-temperature history of the aluminum core support plate. This distinction between shield regions has been retained, even though all calculations are for no-flow cases, because of the widely different nuclear heating rates in the two regions.

The physical constants required for the calculations are the inner and "effective" outer radii of the channel in the three sections with the effective outer radii being obtained as described earlier for other components, the length of each section, the density of aluminum, and the emissivity and shape factors for radiation across the gap between the shield-second pass and the support plate and across the gap between the support plate and the core inlet segment.

The modes of heat transfer considered in calculation of the temperature profiles of these sections are:

1. Nuclear energy deposition in a segment,
2. Axial conduction of heat from (or to) adjacent segments,
3. Radiative heat transfer between the core support plate and the shield second pass, and
4. Radiative heat transfer between the core inlet segment and the core support plate.

At present, a model is in preparation for use in computations of temperature profiles in the projected lithium hydride flight shield.

1.2.1.8 PRESSURE VESSEL DOME MODEL

The pressure vessel dome is divided into sixteen sections. Fifteen of these, starting at the pressure vessel and proceeding to the apex of the dome, are in the form of annular rings. The sixteenth section, at the apex, is a solid piece of material. Calculations of net heat transfer, and temperature changes are made for each section.

These include:

1. Conduction from section to section,
2. Radiation to the atmosphere, and
3. Heat input to the section due to re-entry.

1.2.2 EQUATIONS USED FOR HEAT TRANSFER

1.2.2.1 CONVECTION

No convective heat transfer has been considered since the two conditions assumed are:

1. Coolant flow stoppage and
2. Space environment, i.e., very high vacuum conditions.

1.2.2.2 CONDUCTION

Two cases have been considered for conduction of heat.

1. Axial conduction

$$\frac{\Delta q}{\Delta t} = kA \frac{\Delta T}{\Delta X}$$

where:

$\frac{\Delta q}{\Delta t}$ = rate of heat transfer along the segment being considered during the time interval Δt, k = thermal conductivity of the segment evaluated at its mean temperature during the time interval Δt,

A = cross-sectional area perpendicular to the direction of heat transfer,

ΔX = length of the segment being considered, and

$\frac{\Delta T}{\Delta X}$ = temperature gradient along the segment.

2. Radial conduction through a cylindrical shell

$$\frac{\Delta q}{\Delta t} = \frac{2 \pi L k \Delta T}{\ln (R_2/R_1)}$$

where:

$\frac{\Delta q}{\Delta t}$ = rate of heat transfer through the shell in the radial direction during the time increment Δt,

L = length of the cylindrical shell,

k = thermal conductivity of the shell evaluated at its mean temperature during Δt,

ΔT = temperature difference between the inner and outer surfaces of the shell,

R_2 = radius of the outer surface of the shell, and

R_1 = radius of the inner surface of the shell.

It should be noted that when the shell is very thin, so that $R_1 \cong R_2$, then $R_2/R_1 \cong 1$ and

$$\ln(R_2/R_1) \cong \left(\frac{R_2}{R_1} - 1\right)$$

$$= \frac{R_2 - R_1}{R_1}$$

then the equation for radial heat transfer through a shell can be rewritten as:

$$\frac{\Delta q}{\Delta t} = \frac{(2\pi R_1 L)k\Delta T}{(R_2 - R_1)} = kA'\left(\frac{\Delta T}{\Delta X}\right)'$$

where:

A' = the surface area of the inner surface of the shell,

$\Delta X=(R_2-R_1)$ = the shell thickness, and

$\left(\frac{\Delta T}{\Delta X}\right)'$ = the radial temperature gradient across the shell.

1.2.2.3 RADIATION

The rate of radiative heat transfer between two surfaces a and b can be expressed as: $\Delta q/\Delta t = E\sigma A(T_a^4 - T_b^4)$

where:

$\frac{\Delta q}{\Delta t}$ = the rate of heat transfer between the surfaces during the time increment Δt,

E = a combined surface emissivity and shape factor constant,

A = radiating surface area,

T_a = mean temperature of surface a during Δt,

T_b = mean temperature of surface b during Δt, and

σ = the Stefan-Boltzmann constant.

In all calculations, the value of the Stefan-Boltzmann constant used was:

$$\sigma = 1.713 \times 10^{-9} \text{ BTU/ft}^2\text{-hr-°R} = 3.3044 \times 10^{-15} \text{ BTU/in}^2\text{-sec-°R}$$

1.2.3 LINK II COMPUTER PROGRAM

A brief description of the main program and subroutines contained in LINK II is given below. Flow diagrams for these routines are given in Appendix A.

1.2.3.1 MAIN II

MAIN II is the controlling program for the second link. This program controls the length of time steps for heat transfer purposes and the number of these time steps between decay power calculations in LINK III. MAIN II calls the other subroutines of LINK II which perform heat transfer calculations for the individual reactor components.

1.2.3.2 POWERN

This subroutine calculates the fission power variation with time after shutdown. It also partitions the power deposition among the reactor components.

1.2.3.3 NOZZLE

Performs net heat transfer and temperature calculations for the nozzle tubes and backing. Axial heat conduction along the nozzle backing has been included in the calculations.

1.2.3.4 DOME

Performs heat transfer and temperature calculations for the pressure vessel dome.

1.2.3.5 SETUP

This routine is used whenever temperatures and power levels are transferred from LINK II to LINK III for inventory and diffusion calculations. SETUP reduces the sixteen (16) core sections used in LINK II, for heat transfer purposes, to ten equivalent sections for inventory calculations in LINK III. This is necessary because the large amount of computer storage required to retain the inventory for eighty decay chains does not permit the use of more than ten core sections in LINK III.

1.2.3.6 NCORE

This routine calculates heat transfer and temperature changes for the axial core sections.

1.2.3.7 TIEROD

Calculates net heat transfer and temperature changes for axial sections of the tie-rod, stainless steel sleeve, and pyrographite sleeve.

1.2.3.8 NREFLS

This routine performs heat transfer and temperature calculations for the graphite filler strips, graphite reflector, and beryllium reflector.

1.2.3.9 VESSEL

This routine calculates heat transfer and temperature changes for the pressure vessel.

1.2.3.10 ALSHLD

This routine performs heat transfer and temperature calculations for an aluminum shield made from a solid block of material.

1.2.3.11 SPLATE

Calculates heat transfer and temperature changes for the core support plate.

1.3 LINK III

1.3.1 GENERAL

LINK III performs calculations of the fission product inventory and diffusion for the isotopes of eighty decay chains in each of ten core sections.

This link is first used to simulate the build-up of fission products, and their diffusion, during the normal reactor run prior to loss of coolant. This may be any startup-run-shutdown-restart profile consisting of as many as forty time intervals which may be of any desired length. A power level and temperature is specified for each core section corresponding to each time interval. At the conclusion of any or all of the specified time intervals, the following data may be printed out:

1.) Remaining fission product inventory in each core section and/or total core inventories.

2.) Curies of each isotope which has diffused from each section and/or the total which has diffused from the core.

At the conclusion of the simulation of the normal operating period, the decay power is calculated from the remaining inventory of the individual isotopes and their respective decay energies. A second calculation of the decay power is made at a time equal to half of the first time step in LINK II following loss of coolant. These consecutive values of the decay power are used in

LINK II to establish a rate at which the decay power diminishes. This rate is used to obtain the decay power, by extrapolation, for time intervals in LINK II during which LINK III is not called. Transfer is then made to LINK II which performs heat transfer and temperature calculations for the reactor components until the temperature change in any core section, during the time following the previous decay power calculation, exceeds 400°R. Transfer is then returned to LINK III and the process is repeated.

1.3.2 INVENTORY AND DIFFUSION MODEL

The model used to represent diffusion of fission products from the fuel is based on diffusion rate data obtained at WANL. It has been found that, for fuel samples which have been irradiated at room temperature and are subsequently heated at a temperature (T) for a period of time (t), the fraction of the nuclide produced by the irradiation which has not diffused from the fuel may be given by:

$$G = e^{-Dt} \qquad (1.3.1)$$

where the constant D is obtained from the Arrhenius equation

$$D = D_o e^{-E/RT} \qquad (.3.2)$$

The value of D at an infinitely large temperature, D_o, and the ratio of the activation energy to the gas constant, (E/R), are determined experimentally.[5] Although the diffusion of most nuclides may be represented by a single exponential, some elements require the use of two exponential terms, i.e.

$$G = G_1 e^{-D_1 t} + G_2 e^{-D_2 t} \quad (1.3.3)$$

The diffusion rate for one of these fractions (G_1) is much greater than that for the other fraction. The following rationale of the dual rate process has been offered. If a calculation is made of the fraction of recoil nuclei which would be expected to escape the fuel bead and lodge in the pyrocoat, a value of .15 is obtained. Determinations of G_1 from experimental data yield values in the range 0.1 to 0.2 and, therefore, values of G_2 from 0.8 to 0.9. It therefore seems reasonable to assume that the two release rates are characteristic of the two fractions of recoil nuclei, that which remains in the bead and that which is deposited in the pyrocoat.* The isotopes, which comprise the eighty decay chains in the library used by LINK III, were placed in eight categories according to relative rates of diffusion from fuel which has not been heated to a temperature in excess of 2500°K. Each of these categories is characterized by a different value of D_o and (E/R).[5]

Experimental data indicates that when the fuel temperature reaches 2500°K, rapid degradation of the pyrocoat occurs. In the event that the temperature of a fuel section reaches 2500°K, diffusion coefficients are calculated which are characteristic of release of the isotope of interest from "degraded" fuel. Eleven "degraded-fuel" diffusion categories have been

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* This double-rate effect was observed only for the more rapidly diffusion elements. Although a fraction of all elements would be deposited in the pyrocoat the difference in rates is not measureable except in the case of a few rapid diffusers.

established.

It should be noted that equations (1.3.1) and (1.3.3) are characteristic of diffusion of an isotope with a half-life which is long relative to the heating period. These equations assume no decay and no production of the isotope of interest during the heating period. The calculation of fission product inventories retained and released from a section of an operating reactor required that the rates of the following simultaneous processes be included in the equations used:

(1) Production of the isotope due to the fission process, $PG_i \gamma_i$,

(2) Production by decay of parent nuclides in the decay chain,

$$\sum_{j=1}^{i-1} B_{j \to i} \lambda_j N_j,$$

(3) Loss due to radioactive decay, $\lambda_i N_i$,

(4) Loss due to diffusion, $D_i N_i$,

$$\frac{dN_i}{dt} = PG_i \gamma_i - \lambda_i N_i - D_i N_i + \sum_{j=1}^{i-1} B_{j \to i} \lambda_j N_j \tag{1.3.4}$$

and

$$\frac{dL_i}{dt} = D_i N_i \tag{1.3.5}$$

where:

N_i = number of atoms of isotope (i) in the given core section,

P = power level in the core section (fissions/second),

γ_i = fission yield of isotope (i),

λ_j = decay constant of isotope (j) which is a precursor of isotope (i),

$B_{j\,i}$ = branching ratio for the production of isotope (i) from a precursor (j),

D_i = diffusion constant (characteristic of the isotope and the temperature of the core section),

G_i = is a factor which is used to account for the faster diffusion of the fraction of some nuclides which are deposited external to the fuel beads by recoil,

L_i = number of atoms of isotope (i) which have diffused from the core section.

There are several assumptions which are inherent in the use of the above equations. Two of the most important of these are:

(1) The fractional release rate for any element is independent of the size of the fuel sample. The experimental rates were obtained using fuel samples 0.25" in diameter x 0.25" in length, and these have have been used in attempting to predict the release from full size core sections with a radius of 17.8' and a thickness of five inches. This assumption has been validated to some degree by a series of experiments in which the relative release rates were measured for

different size samples ranging from cylinders of 1/16" length to full size elements 52" long. No measureable variation in the relative release rate was observed with size of the fuel sample.

(2) The nuclei which diffuse from one core section do not become deposited in a cooler section.

There is no direct evidence as to the validity of the second assumption. However, a comparison has been made of results obtained, using FIPDIF[1] to predict release fractions during the NRX-A2 runs with measured values after the run. This comparison indicates that there is good agreement and that this assumption is valid.

1.3.3 LINK III COMPUTER PROGRAM

LINK III contains a main program and two subroutines. A brief discussion of the operations performed by these routines is given below. Flow diagrams illustrating the sequence of these operations are given in Appendix A.

1.3.3.1 MAIN III

MAIN III is the controlling program for LINK III. In this program, the axial temperature and power profile are used to calculate changes in inventory and diffusion for each core section during a specified time interval. For each section, diffusion coefficients are evaluated which are characteristic of the diffusion group and the temperature of the core section. Removal coefficients are formed for each isotope which are the sum of the diffusion coefficient and the decay constant for the isotope. Subroutine EXINT is then called to evaluate the successive integrals required in the solution of equations (1.3.4) and (1.3.5). After calculations are completed for a given section, the inventory of each isotope which has diffused from that section is stored on magnetic tape. After completion of calculations for all sections, the fission product inventory is written on the output tape if output has been requested at this time in the simulation. This is accomplished by the FIPOUT subroutine. The axial temperature and power shape corresponding to the next time increment are then used, and this process is repeated until the simulation of the input operating profile has been completed.

After loss of coolant, the calculations performed by LINK III involve only one pass through the time-step loop. The length of the time step is the length of real time for which the simulation has been carried out in LINK II subsequent to the previous inventory calculations in LINK III. The axial temperature and power shape for these calculations are time-weighted values supplied by LINK II.

1.3.3.2 EXINT

EXINT evaluates the successive integrals referred to above.

1.3.3.3 FIPOUT

FIPOUT writes inventory data on the output tape at any real-time point in the simulation at which output has been requested. At the conclusion of the simulation, the diffusion data for each time increment, which had been written on an intermediate tape, are read back into core. The quantity of any isotope which diffuses from a given core section during a given time increment is added to that which had diffused from the core section during previous time steps. These totals are then written on the output tape at the output times previously requested.

2.0 RESULTS

2.1 SIGNIFICANCE OF NOFLOW-FIPDIF COUPLING

A deficiency of the NOFLOW program was its failure to consider fission product diffusion due to the high temperature achieved in the core following loss of cooling accident. It was thought that inclusion of fission product diffusion would have two major effects:

1.) It would give a more realistic estimate of temperatures of reactor components which might be subjected to thermal degradation in the event of loss of coolant accident.

2.) It would reduce the predicted core inventory and, hence, the predicted potential radiological doses associated with a spent engine.

In order to evaluate the magnitude of the first of these effects, a comparison was made between the temperatures obtained from a NOFLOW computer run with those obtained from a NOFLOW-FIPDIF run. The particular case chosen for analysis was one in which the reactor was operated 613 seconds before failure due to loss of coolant. This operating time includes a 60 second ramp to full power. A two per cent reactivity shutdown was assumed.

Figure 2.1 shows the fuel temperature at the core midplane as a function of time after loss of coolant. This particular core section was selected for examination since it represents the hottest part of the core. As [redacted] Act - 1954

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MISSION MODEL 4 - 613 SEC. POWER OPN. 6-30-65
TEMPERATURE OF FUEL CLUSTERS AT CORE MIDPLANE

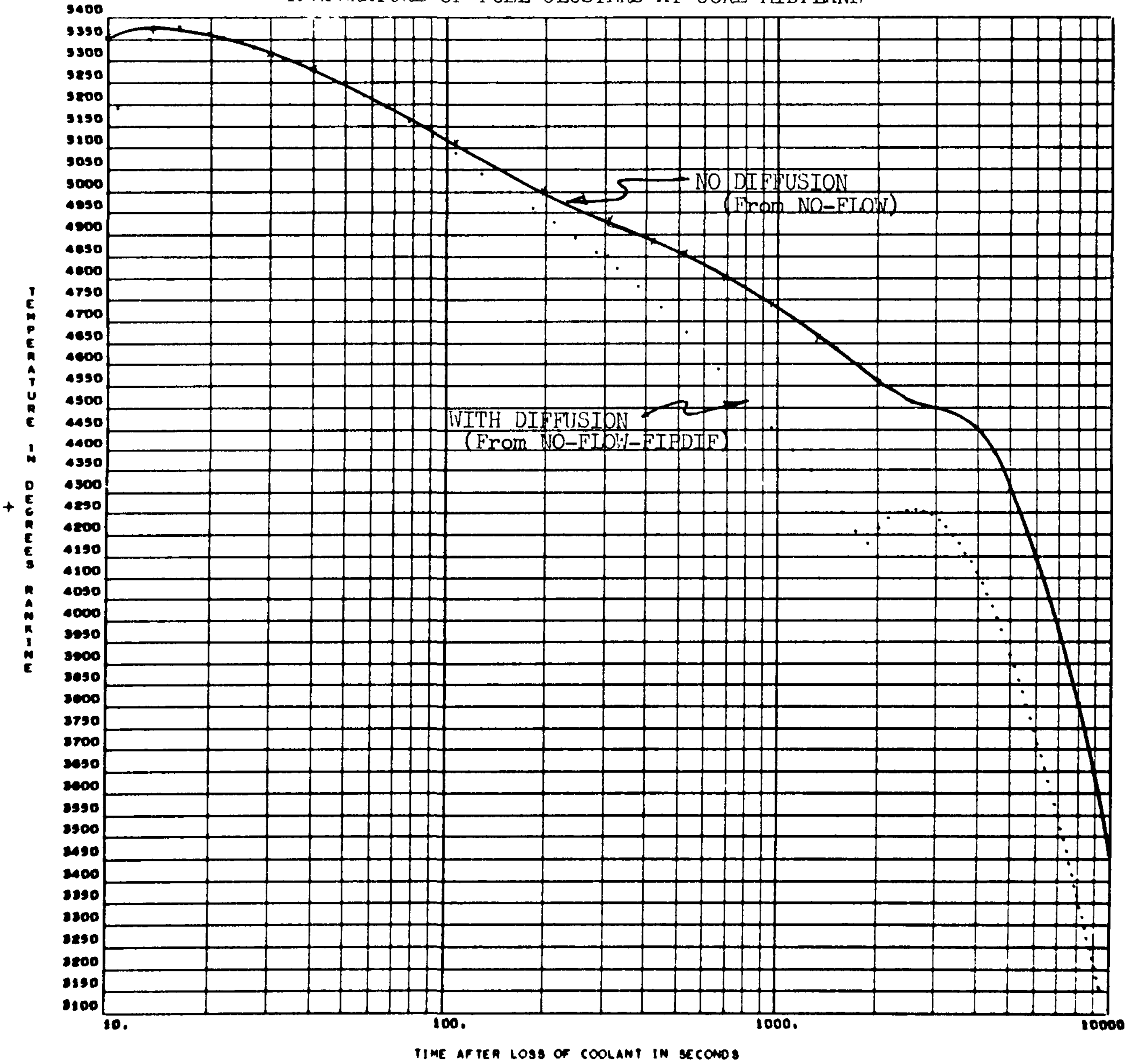

FIGURE 2.1

FUEL TEMPERATURE AT CORE MIDPLANE VS TIME AFTER LOSS OF COOLANT

[redacted]

may be seen from an examination of this figure, diffusion has no influence on core temperature during the first hundred seconds after loss of coolant. This result is to be expected since the primary heat source immediately after shutdown is the decaying neutron population. However, after this time the effect of diffusion on temperature becomes evident with a maximum reduction of 400°R at about 2000 seconds after shutdown.

A number of curves have been included to show the effect of fission product diffusion on the temperature distribution in various reactor components at a time of 4860 seconds after loss of coolant. By this time the temperatures of the outer components are high enough that the effect of reduction of the decay heat source is readily noted.

The following is a list of those figures which illustrate the effect of fission product diffusion on reactor component temperatures:

Figure	Description
2.2	Core Temperature vs Distance From Dome End of Core
2.3	Graphite Reflector Temperature vs Distance From Nozzle End
2.4	Beryllium Reflector Temperature vs Distance From Nozzle End
2.5	Pressure Vessel Temperature vs Distance From Nozzle End

An examination of these figures indicates that at the core mid-plane, diffusion lowers component temperatures by the following amounts: core 400°R;

graphite reflector 250°R; beryllium reflector 350°R; pressure vessel 350°R.

The reduction in fission product inventory caused by diffusion is shown in Table 2.1 for the same reactor operating period. Here the inventories of 18 isotopes are shown. These isotopes were selected for inclusion since they represent species for which the diffusion rates range from very fast to very slow.

In the Table, Column I represents the quantity of each isotope remaining at 5436 seconds after loss of coolant as calculated by the NOFLOW-FIPDIF computer program. Column II represents the quantity which would remain at this time if decay had been the only loss mechanism considered for each isotope. The latter numbers have been calculated with the FPIP computer program. The ratio of these computed inventories for each isotope demonstrates the significance of the effect of diffusion on total core inventory.

A time of 5436 seconds after loss of coolant was selected for examination since at this time the average core temperature is low enough ($< 3400°R$) that further depletion of isotopes by diffusion is negligible. Examination of the data in this table indicated that the ratio of inventory with diffusion to inventory without diffusion varies from 1.0 (no significant loss) to 0.0002.

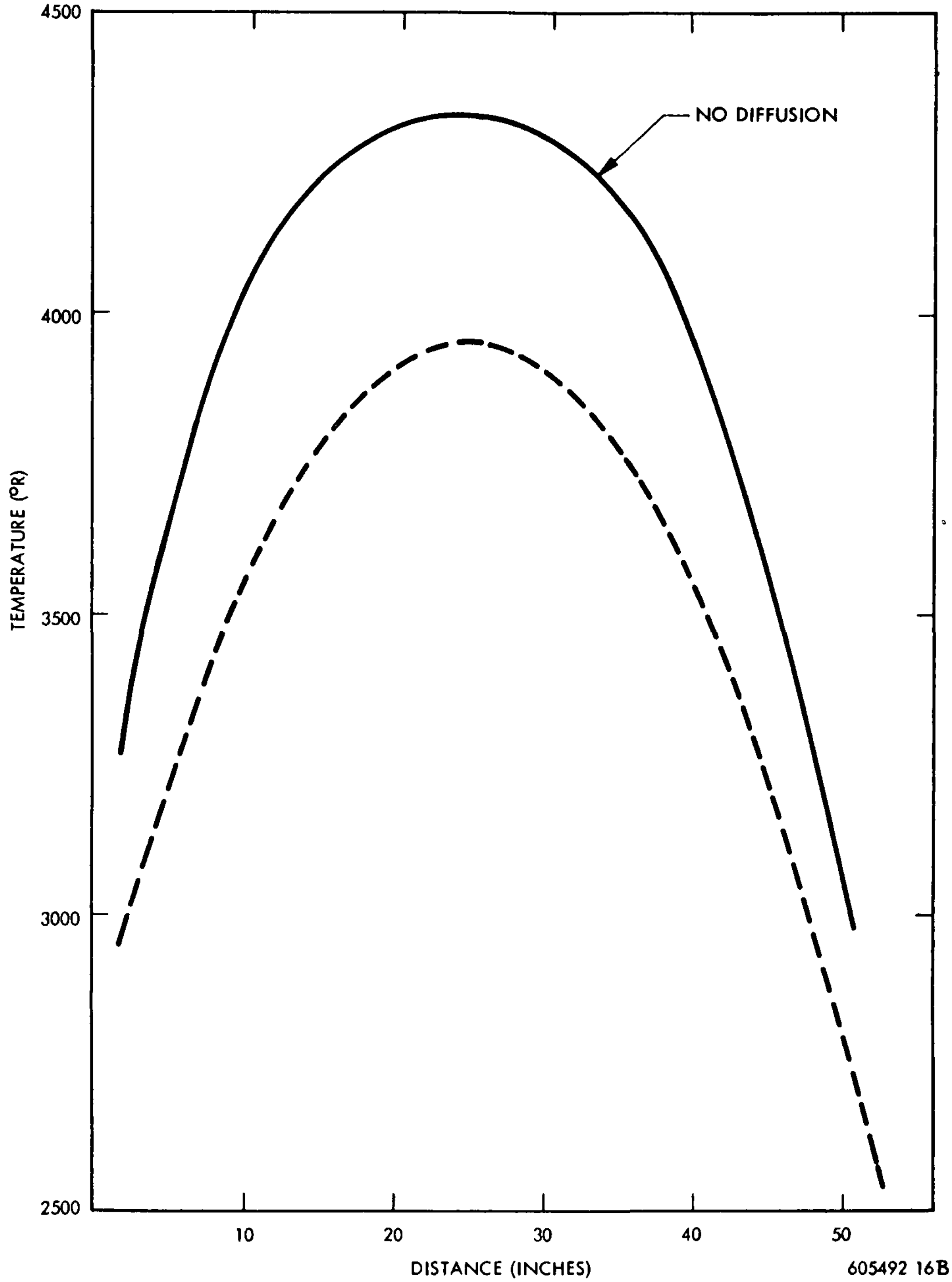

FIGURE 2.2

CORE TEMPERATURE VS DISTANCE FROM DOME END OF CORE

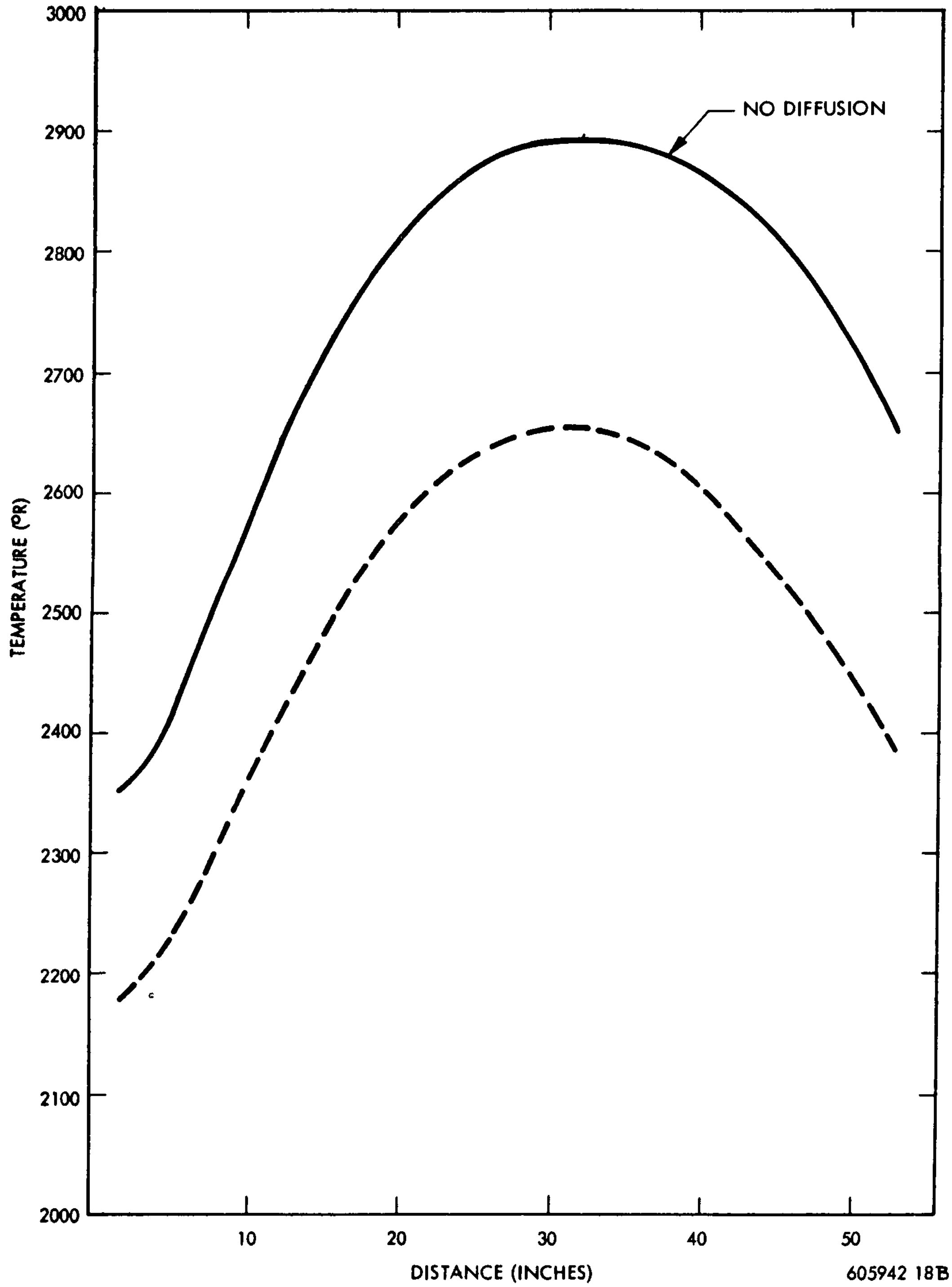

FIGURE 2.3

GRAPHITE REFLECTOR TEMPERATURE VS DISTANCE FROM NOZZLE END

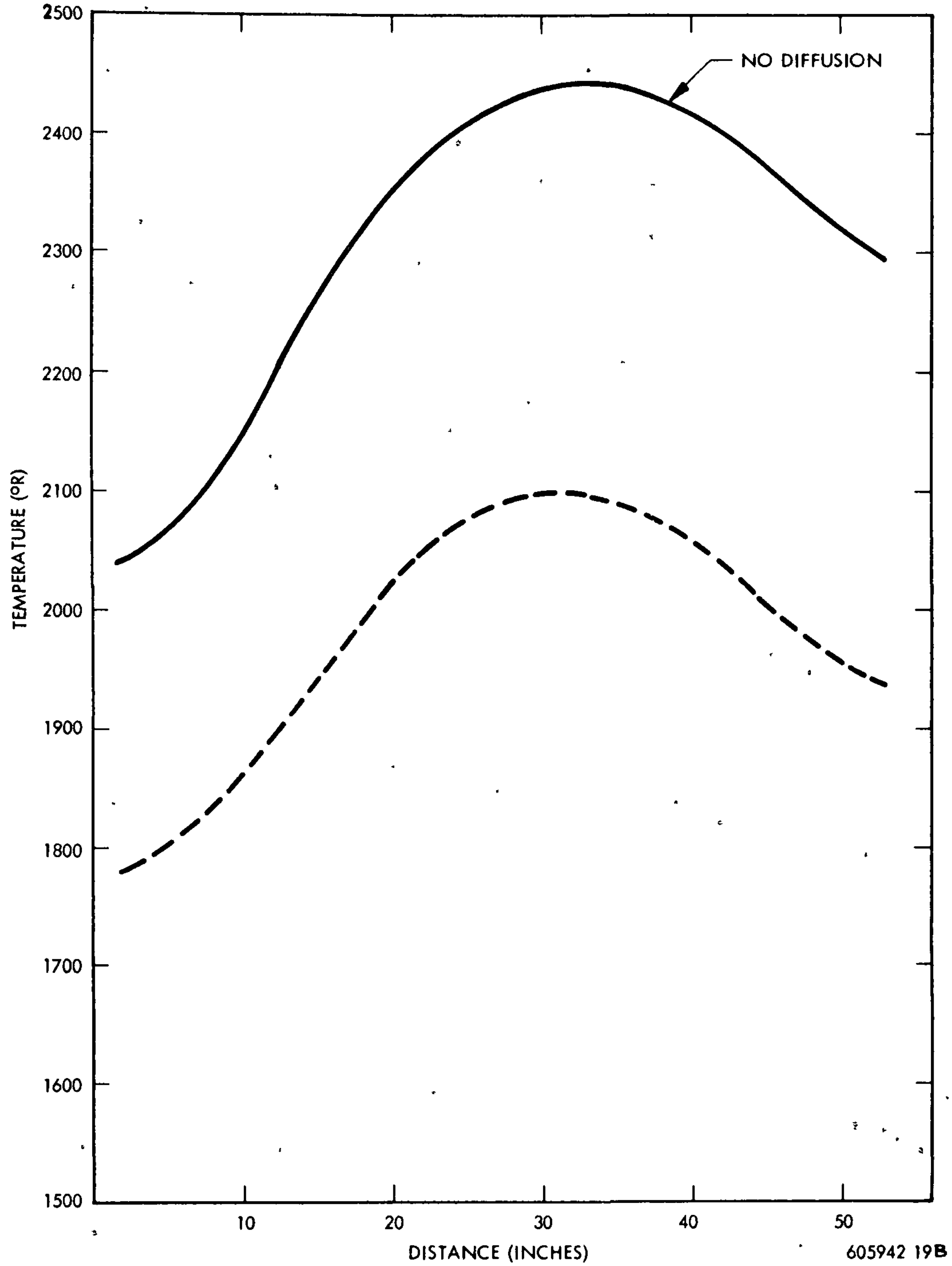

FIGURE 2.4

BERYLLIUM REFLECTOR TEMPERATURE VS DISTANCE FROM NOZZLE END

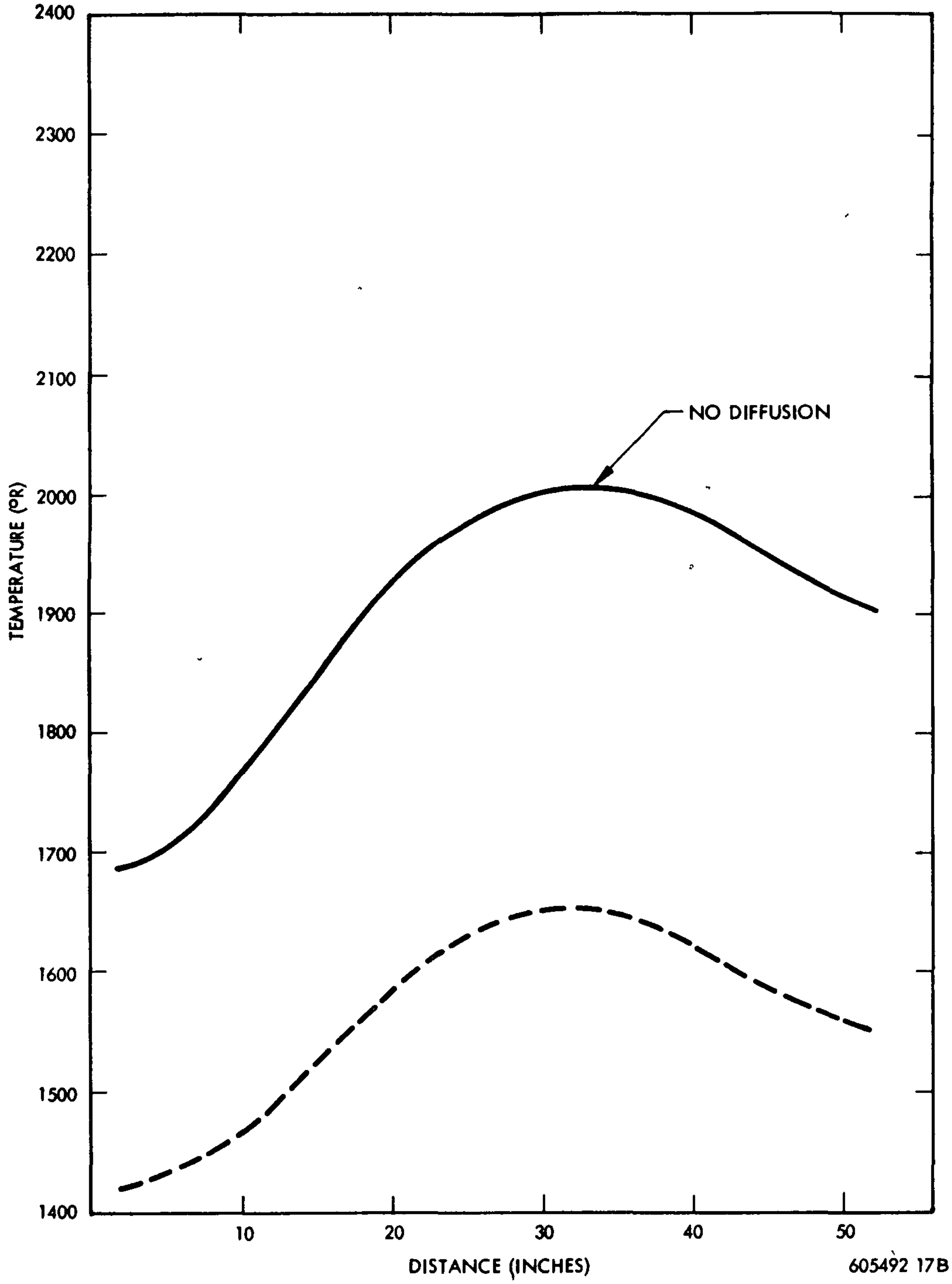

FIGURE 2.5

PRESSURE VESSEL TEMPERATURE VS DISTANCE FROM NOZZLE END

TABLE 2.1

INVENTORY OF SELECTED FISSION PRODUCTS
5436 SECONDS AFTER LOSS OF COOLANT

ISOTOPE	I CURIES REMAINING (With Diffusion)	II CURIES REMAINING (No Diffusion)	RATIO(Col. I/Col. II)
BR-89	0.0	0.0	--
KR-89	0.25×10^{-2}	0.11	0.0227
RB-89	0.3634×10^{5}	0.3267×10^{6}	0.1112
SR-89	0.8851×10^{2}	0.396×10^{4}	0.022
PD-111m	0.3448×10^{2}	0.345×10^{2}	1.0
PD-111	0.2974×10^{4}	0.297×10^{4}	1.0
AG-111m	0.8626×10^{2}	0.157×10^{4}	0.055
AG-111	0.217×10^{2}	0.101×10^{3}	0.215
CD-121	0.0	0.0	--
IN-121m	0.2677×10^{-4}	0.114×10^{-3}	0.235
IN-121	0.0	0.0	--
SN-121m	0.0663	0.1776	0.373
SN-121	0.5630×10^{-1}	0.277×10^{3}	0.0002
SN-131	0.2237×10^{-2}	0.096	0.0233
SB-131	0.3649×10^{5}	0.436×10^{6}	0.0837
TE-131m	0.3412×10^{4}	0.2066×10^{5}	0.1652
TE-131	0.1253×10^{6}	0.109×10^{7}	0.115
I-131	0.831×10^{3}	0.104×10^{5}	0.08

2.2 CASES EXAMINED

The program described in this report has been used to evaluate the consequences to reactor materials of loss-of-coolant accidents following various reactor run profiles. These profiles correspond to cases in which:

1.) The loss of coolant accident occurs subsequent to a sub-orbit start but prior to the time at which the rocket reaches its orbit,

2.) The accident occurs while the reactor is operated the first time while in orbit, or,

3.) The accident occurs during a re-start of the reactor while in orbit.

Four failure times were considered for the cases involving a single firing of the nuclear stage. The failure times selected were 63, 263, 463, and 613 seconds after start-up. Each operating period included a 60 second ramp to full power. It was assumed that the accident, which occurred at one of these times, was accompanied by a reactivity loss such that the reactor was shutdown by two percent.

In the simulation of the cases involving re-start of the nuclear stage, the following power profile was represented:

1.) Start-up ramp -- 60 seconds duration

2.) Full power operation -- 522 seconds duration

3.) Shut-down ramp -- 100 seconds duration

4.) Shut-down (zero power) -- 5288 seconds duration

5.) Re-start ramp -- 40 seconds duration

6.) Full power operation following the re-start ramp

The following four failure times during the re-start period were considered: 63 seconds, 245 seconds, 463 seconds, and 613 seconds.

2.2.1 FAILURE DURING INITIAL FIRING

2.2.1.1 ORBITAL START CASES

Analyses, based on the results of simulations of loss of coolant accidents during the initial firing of the nuclear stage, have assumed that the rocket was in a 100-nautical-mile orbit at the time of start-up. They have also assumed that disassembly of the core into individual fuel elements would occur at a time when either:(6)

1.) the lateral support system fails (the temperature of the graphite reflector at the nozzle end reaches 1260°R - 1680°R), or

2.) the pressure vessel temperature reaches 1680°R at some location causing loss of the part of the pressure vessel between the melted section and the nozzle. This would also result in loss of the nozzle.

Sections 2.2.1.1.1 and 2.2.1.1.2 contain a series of curves and tables which illustrate the type of results which were obtained.

They correspond to a loss of coolant accident following a single period of reactor operation of 613 seconds duration.

2.2.1.1.1 AXIAL TEMPERATURE DISTRIBUTION IN REACTOR COMPONENTS 4860 SECONDS AFTER LOSS OF COOLANT

This section contains a series of curves of the axial temperature distribution in reactor components corresponding to a time 4860 seconds after loss of coolant. They are presented in the following order:

Figure	Description
2.6	Core Temperature vs Distance From Dome End
2.7	Filler Strip Temperature vs Distance From Nozzle End
2.8	Graphite Reflector Temperature vs Distance From Nozzle End
2.9	Beryllium Reflector Temperature vs Distance From Nozzle End
2.10	Pressure Vessel Temperature vs Distance From Nozzle End
2.11	Shield 1st Pass Temperature vs Distance From Nozzle End
2.12	Shield 2nd Pass Temperature vs Distance From Nozzle End
2.13	Support Plate Temperature vs Distance From Core End

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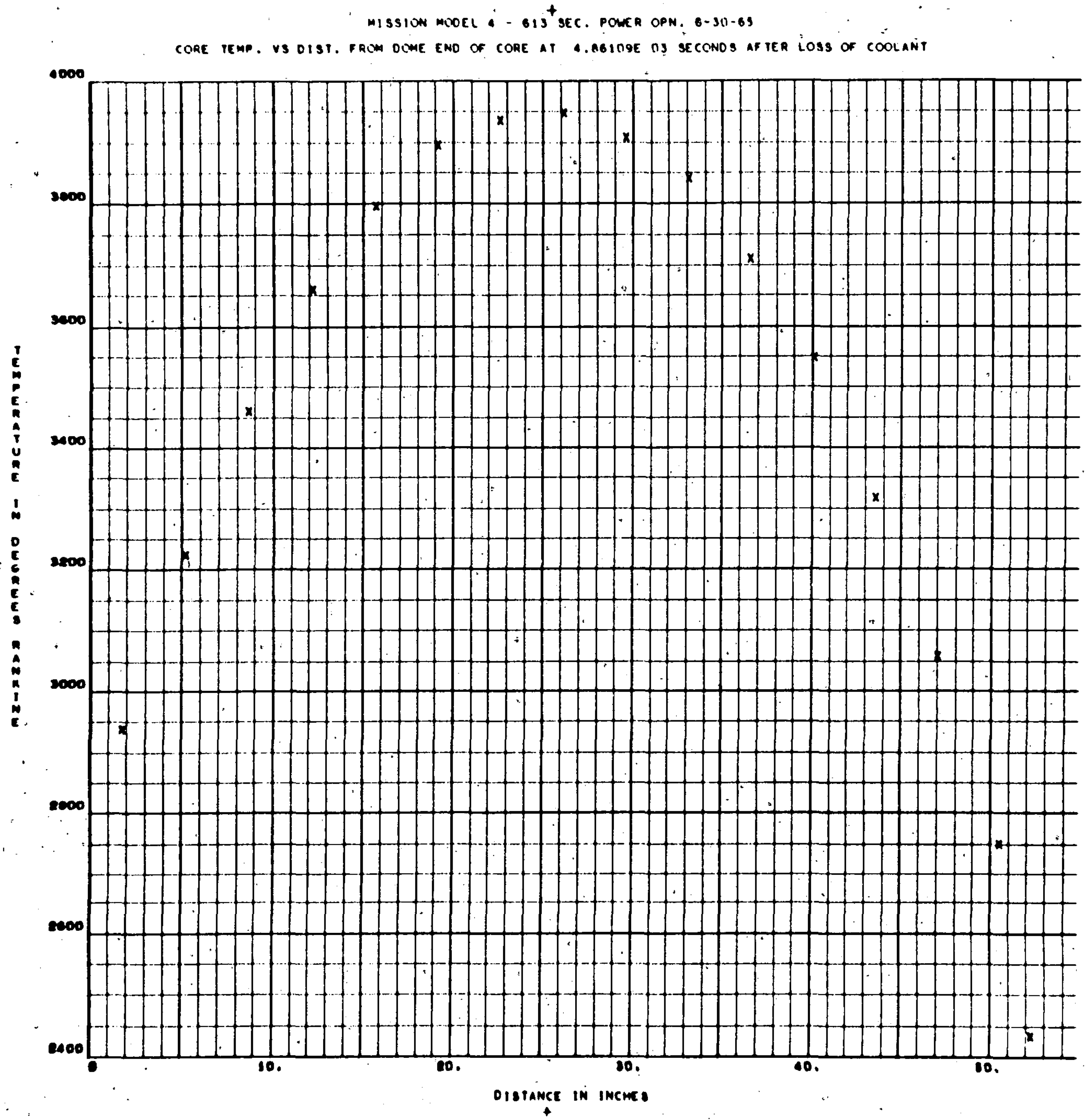

FIGURE 2.6

613 SECOND REACTOR OPERATION

CORE TEMPERATURE VS DISTANCE FROM DOME END

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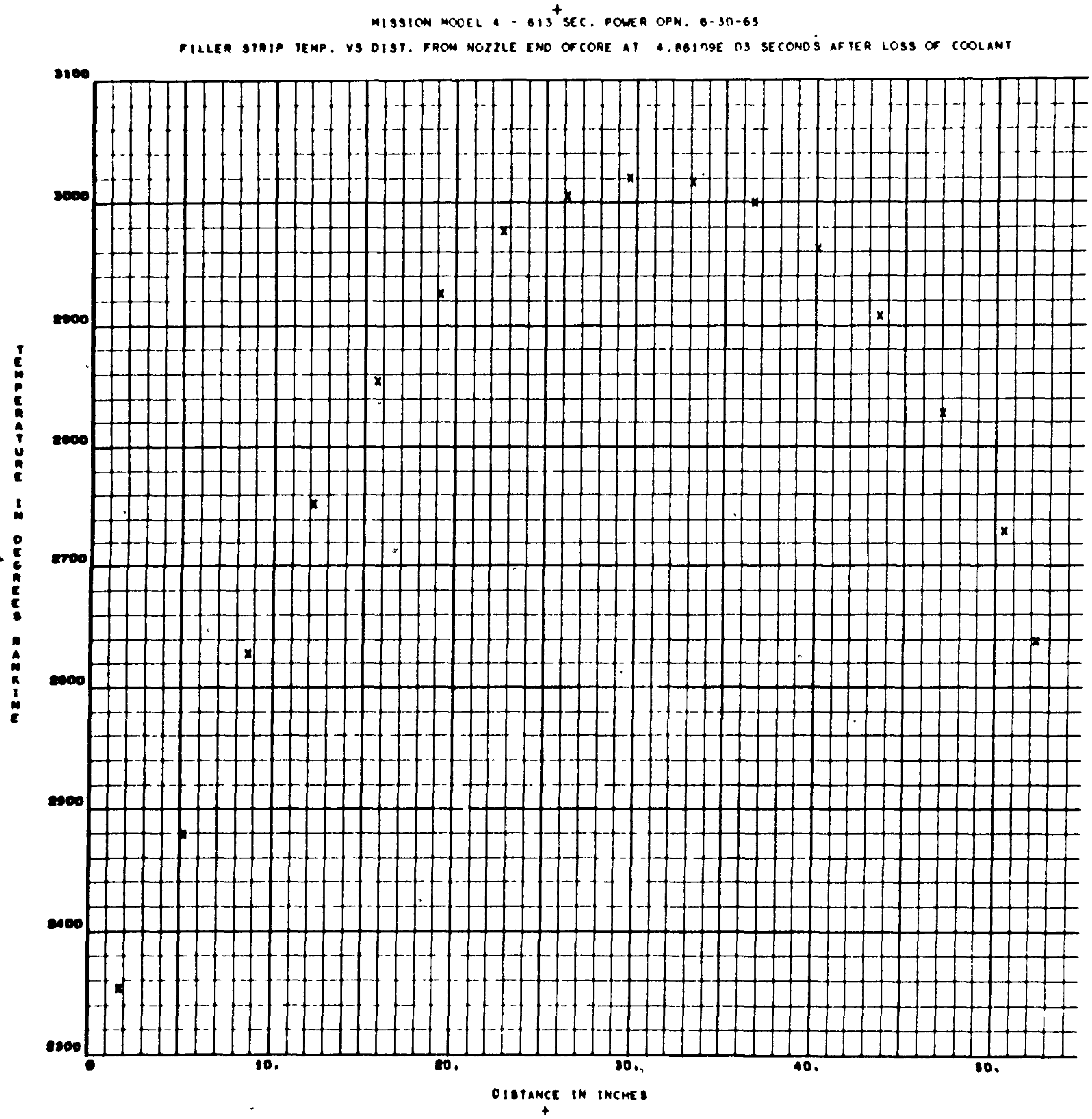

FIGURE 2.7

613 SECOND REACTOR OPERATION

FILLER STRIP TEMPERATURE VS DISTANCE FROM NOZZLE END

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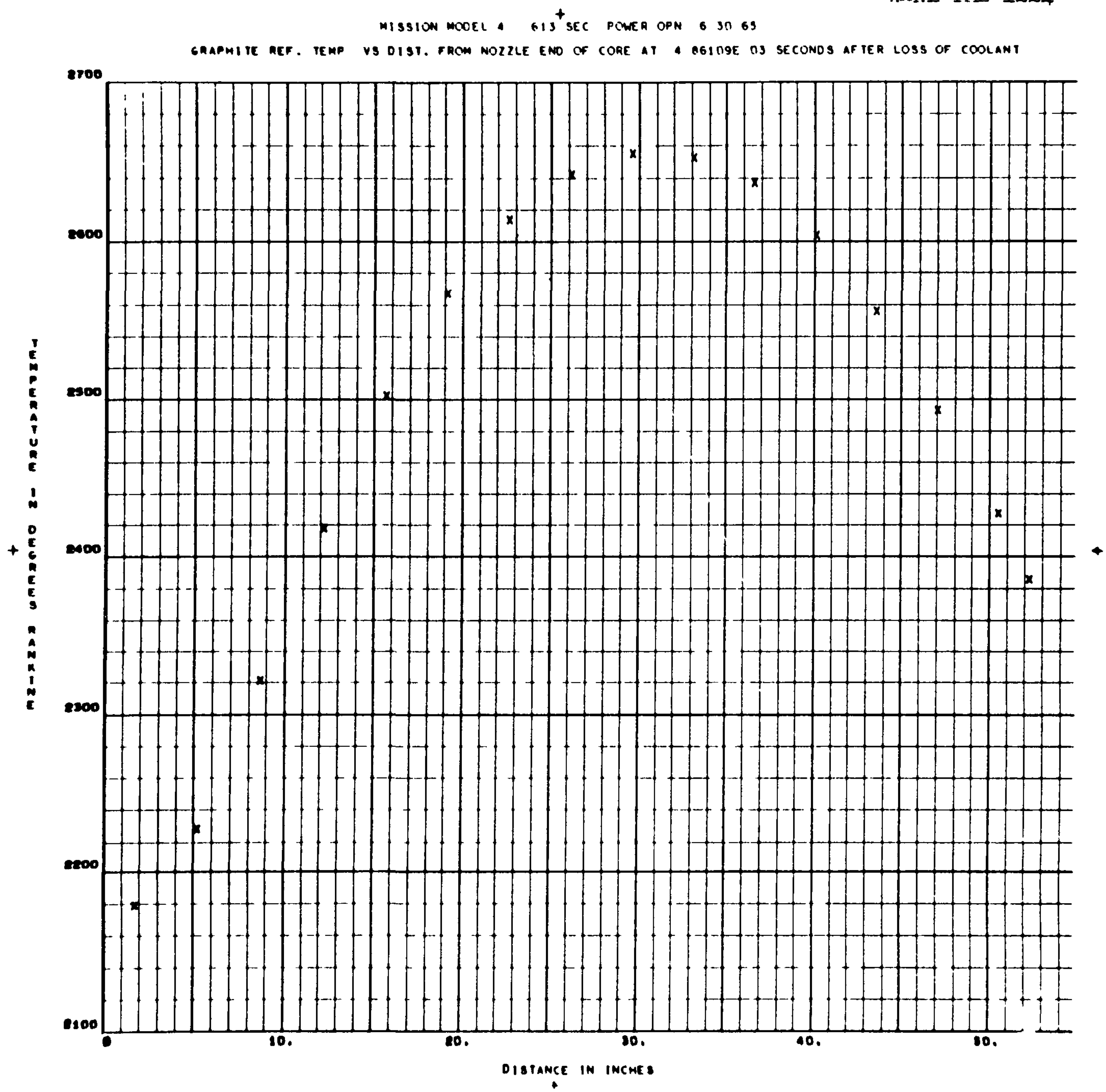

FIGURE 2.8

613 SECOND REACTOR OPERATION

GRAPHITE REFLECTOR TEMPERATURE VS DISTANCE FROM NOZZLE END

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BE REF. TEMP. VS DIST. FROM NOZZLE END OF CORE AT 4.86109E 03 SECONDS AFTER LOSS OF COOLANT

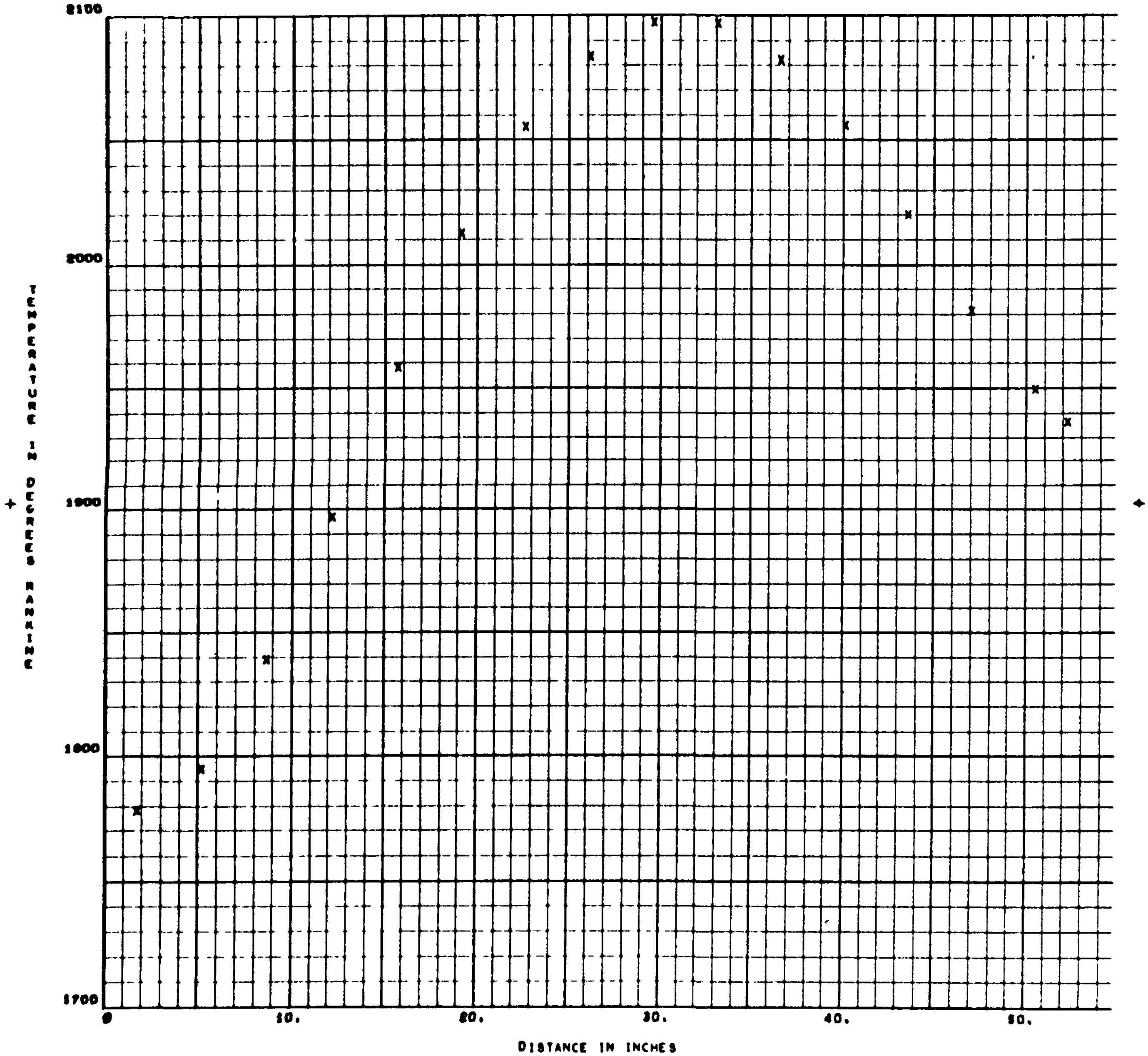

FIGURE 2.9

613 SECOND REACTOR OPERATION

BERYLLIUM REFLECTOR TEMPERATURE VS DISTANCE FROM NOZZLE END

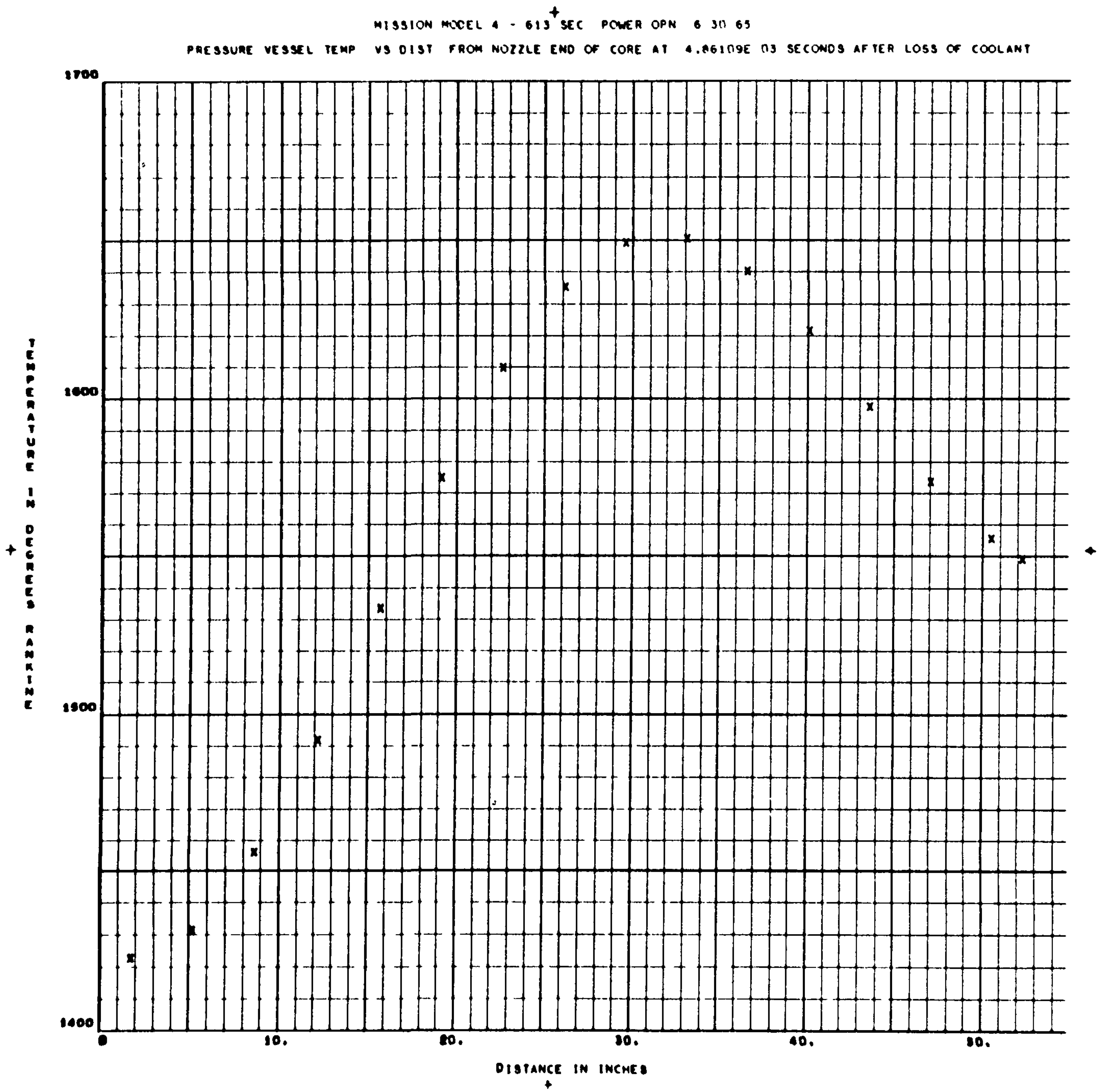

FIGURE 2.10

613 SECOND REACTOR OPERATION

PRESSURE VESSEL TEMPERATURE VS DISTANCE FROM NOZZLE END

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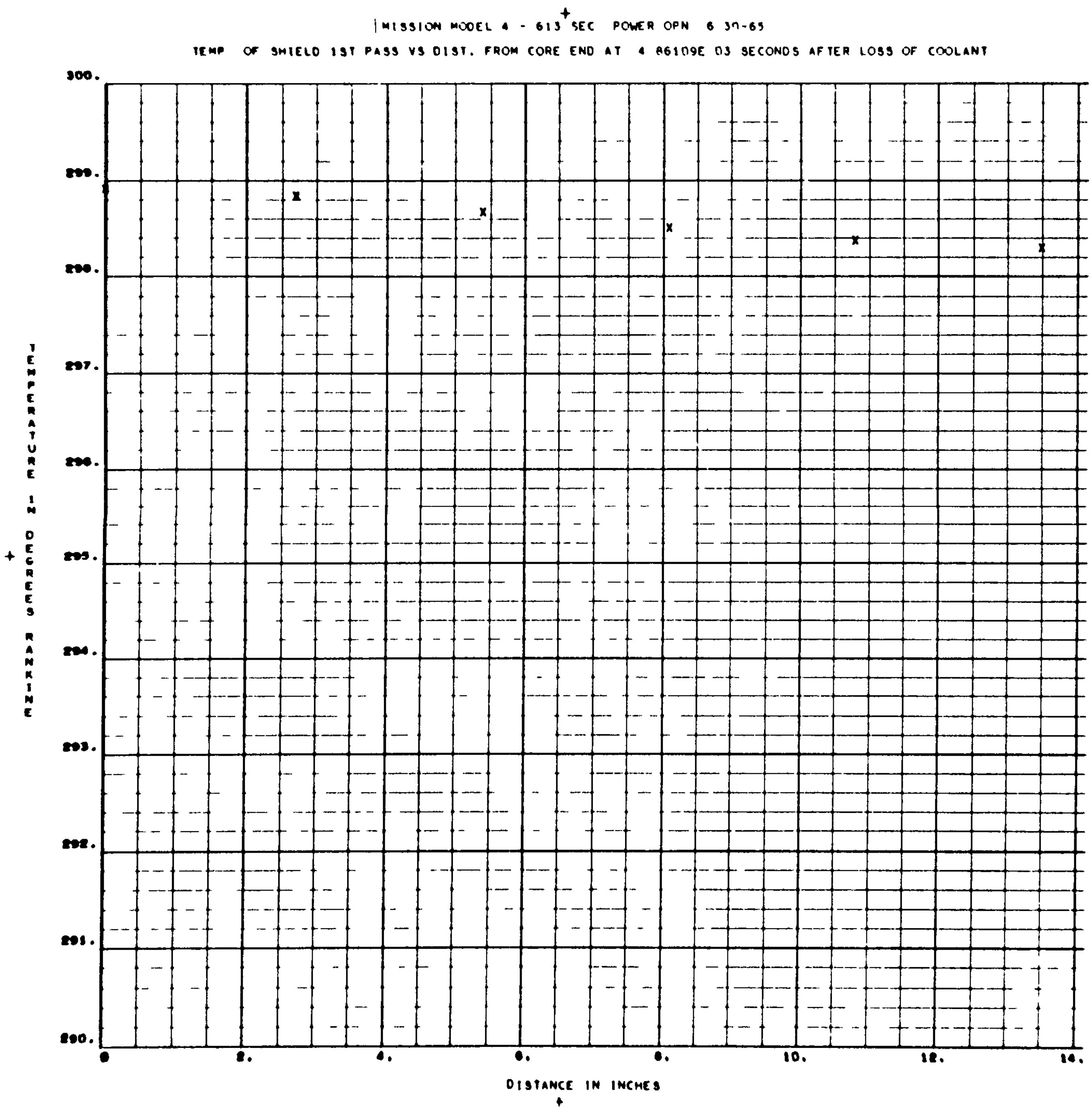

FIGURE 2.11

613 SECOND REACTOR OPERATION

SHIELD 1ST PASS TEMPERATURE VS DISTANCE FROM NOZZLE END

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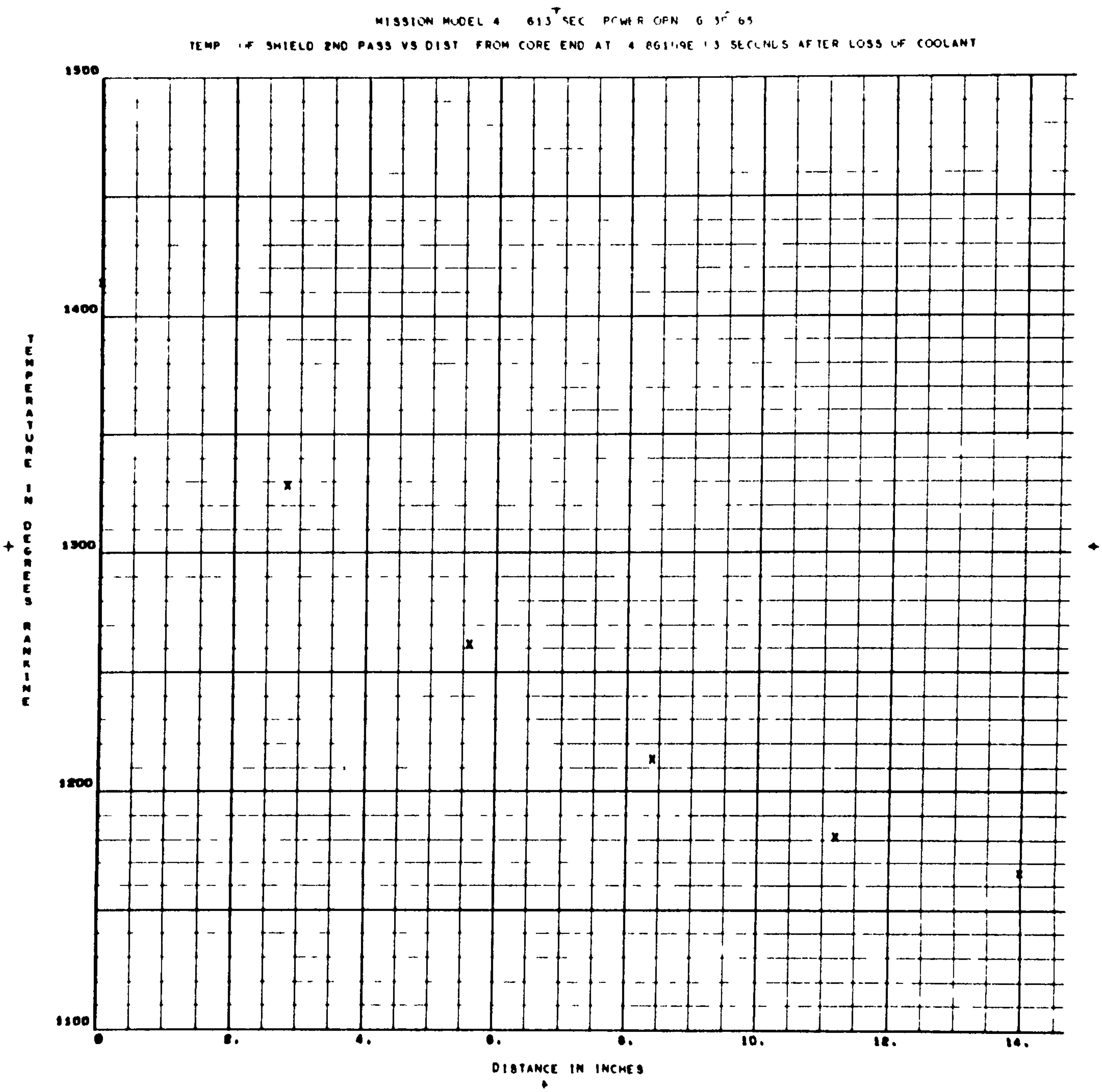

FIGURE 2.12

613 SECOND REACTOR OPERATION

SHIELD 2ND PASS TEMPERATURE VS DISTANCE FROM NOZZLE END

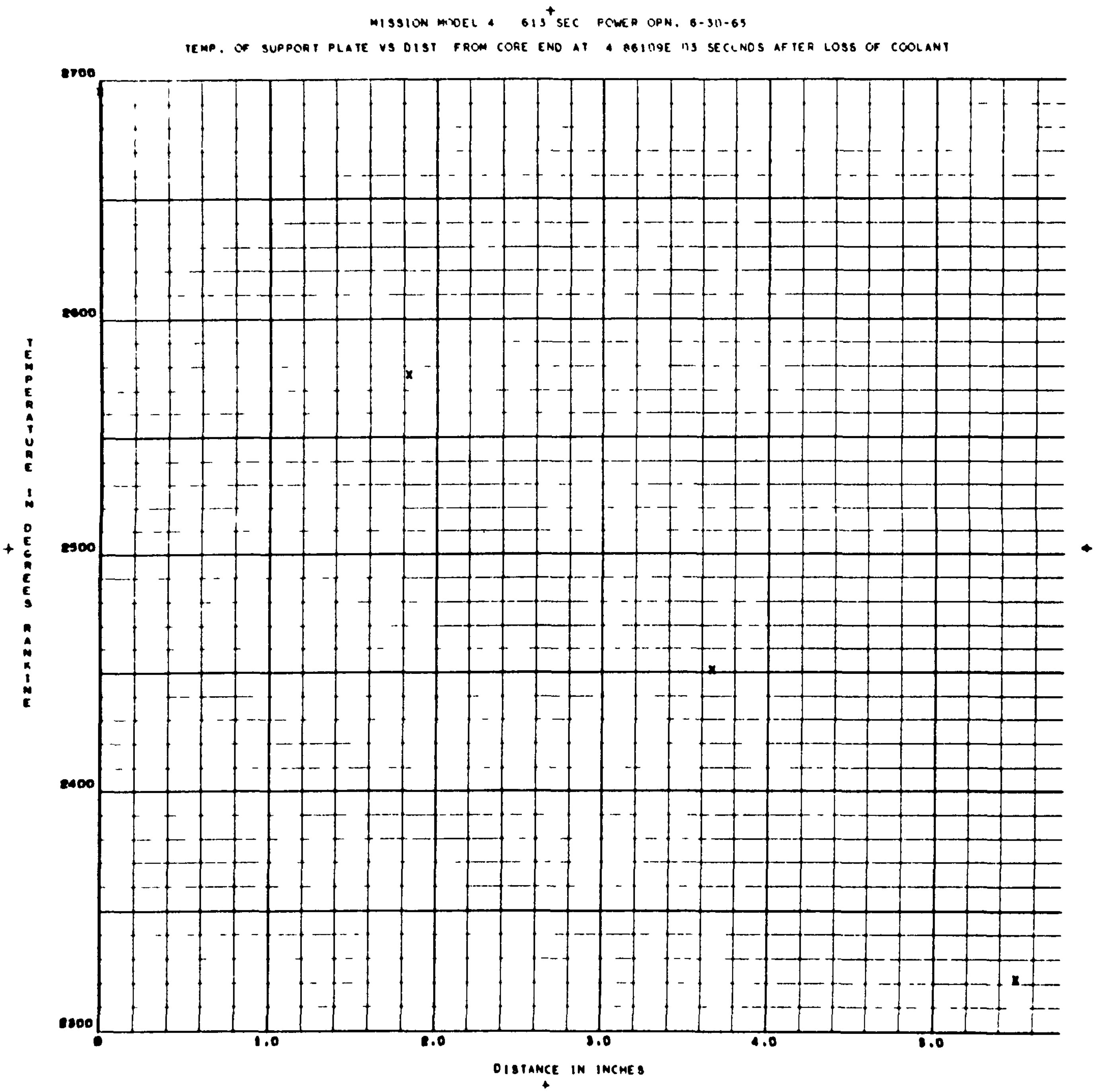

FIGURE 2.13

613 SECOND REACTOR OPERATION

SUPPORT PLATE TEMPERATURE VS DISTANCE FROM CORE END

2.2.1.1.2 TEMPERATURE VS TIME AFTER LOSS OF COOLANT AT SELECTED LOCATIONS IN REACTOR COMPONENTS

A series of curves are presented in this section which illustrate the temperature at one location of each reactor component vs time after loss of coolant occurs.

A list of these figures is given below:

Figure	Description
2.14	Average Core Temperature
2.15	Temperature of Fuel Clusters at Dome End of Core
2.16	Temperature of Fuel Clusters at Nozzle End
2.17	Temperature of Graphite Reflector at Nozzle End
2.18	Temperature of Graphite Reflector at Core Midplane
2.19	Temperature of Graphite Reflector at Dome End
2.20	Temperature of Beryllium Reflector at Core Midplane
2.21	Temperature of Pressure Vessel at Nozzle End
2.22	Temperature of Pressure Vessel at Core Midplane
2.23	Temperature of Center of Core Support Plate

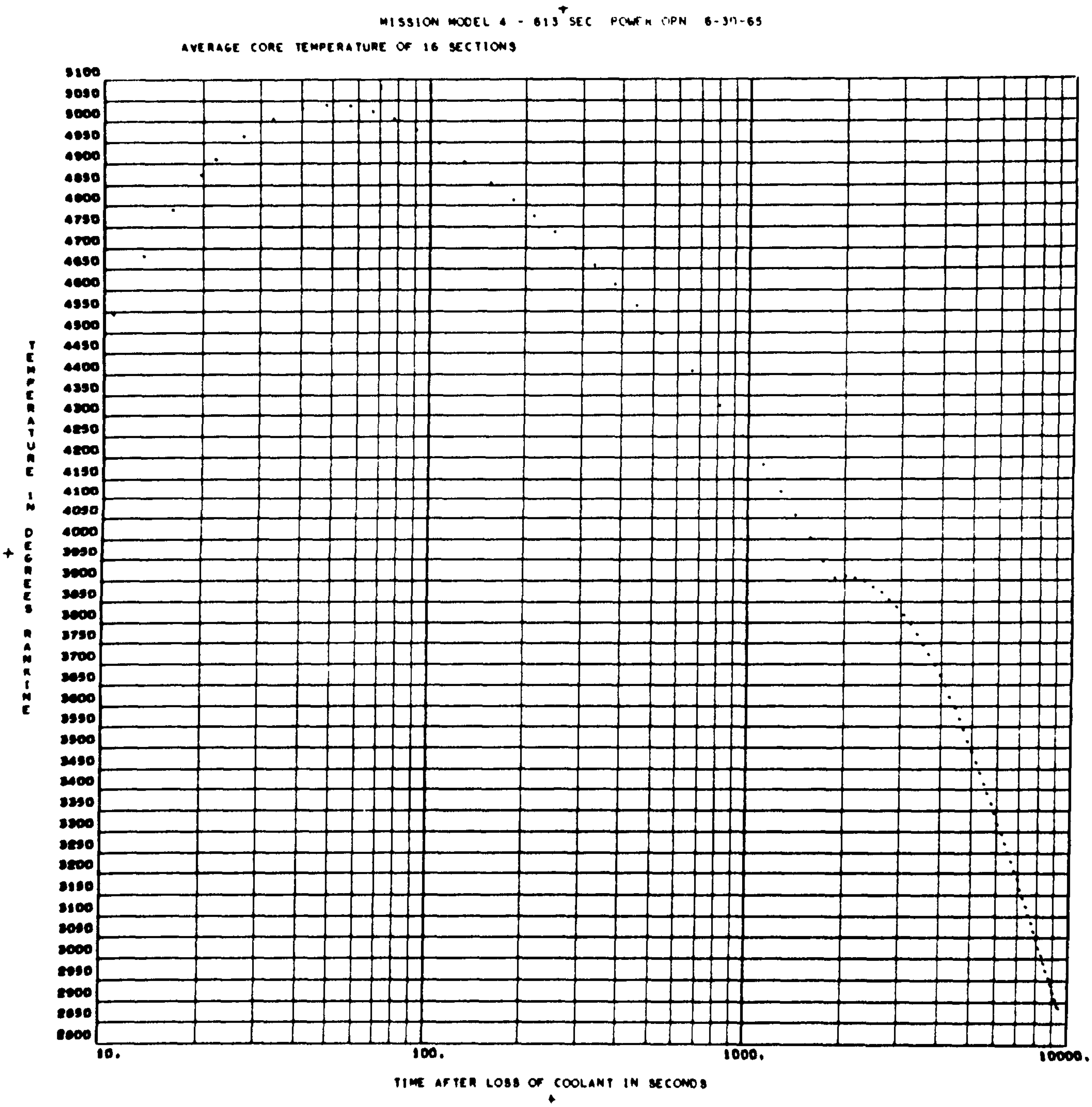

FIGURE 2.14

613-SECOND REACTOR OPERATION

AVERAGE CORE TEMPERATURE

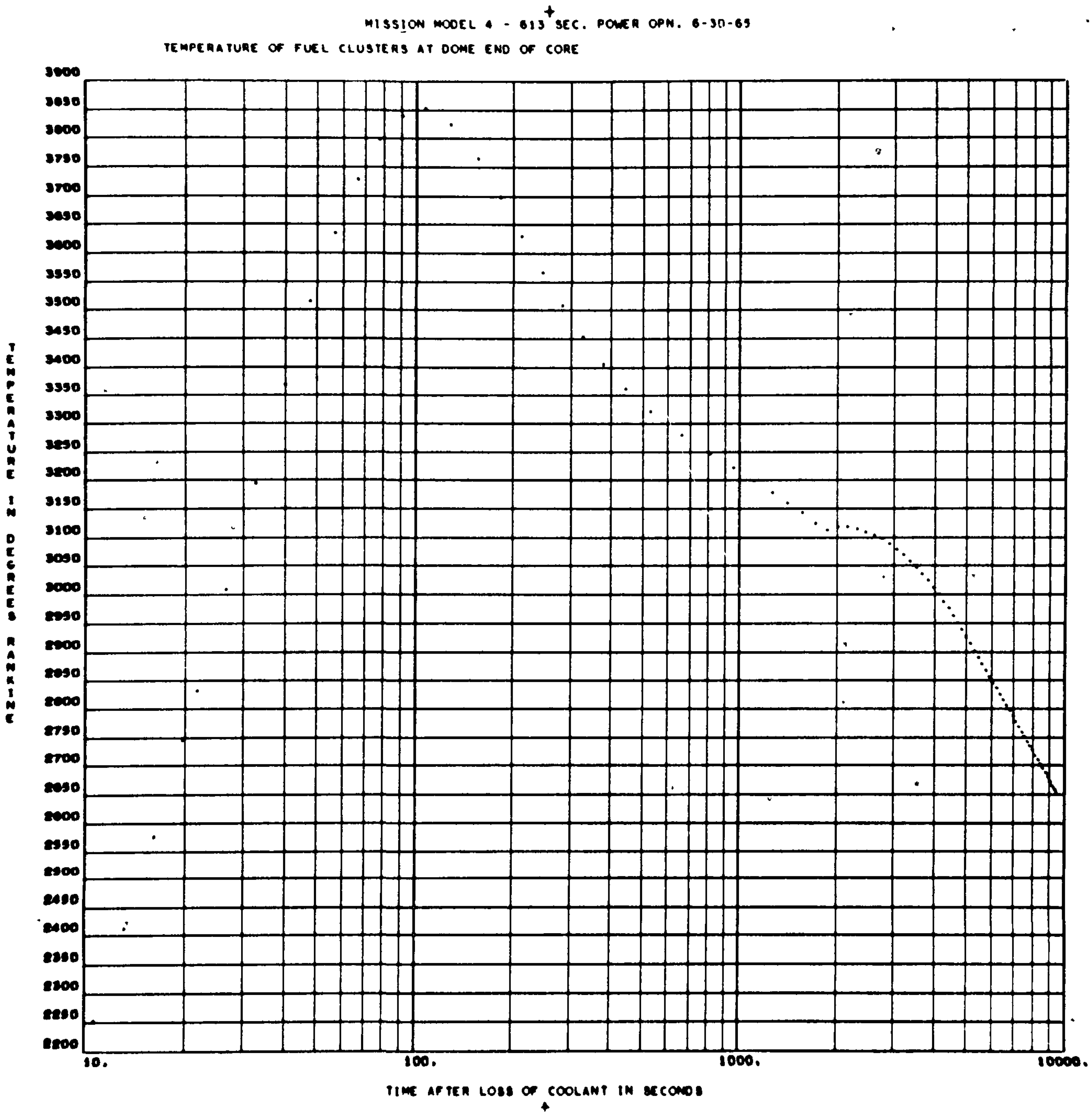

FIGURE 2.15

613 SECOND REACTOR OPERATION

TEMPERATURE OF FUEL CLUSTERS AT DOME END OF CORE

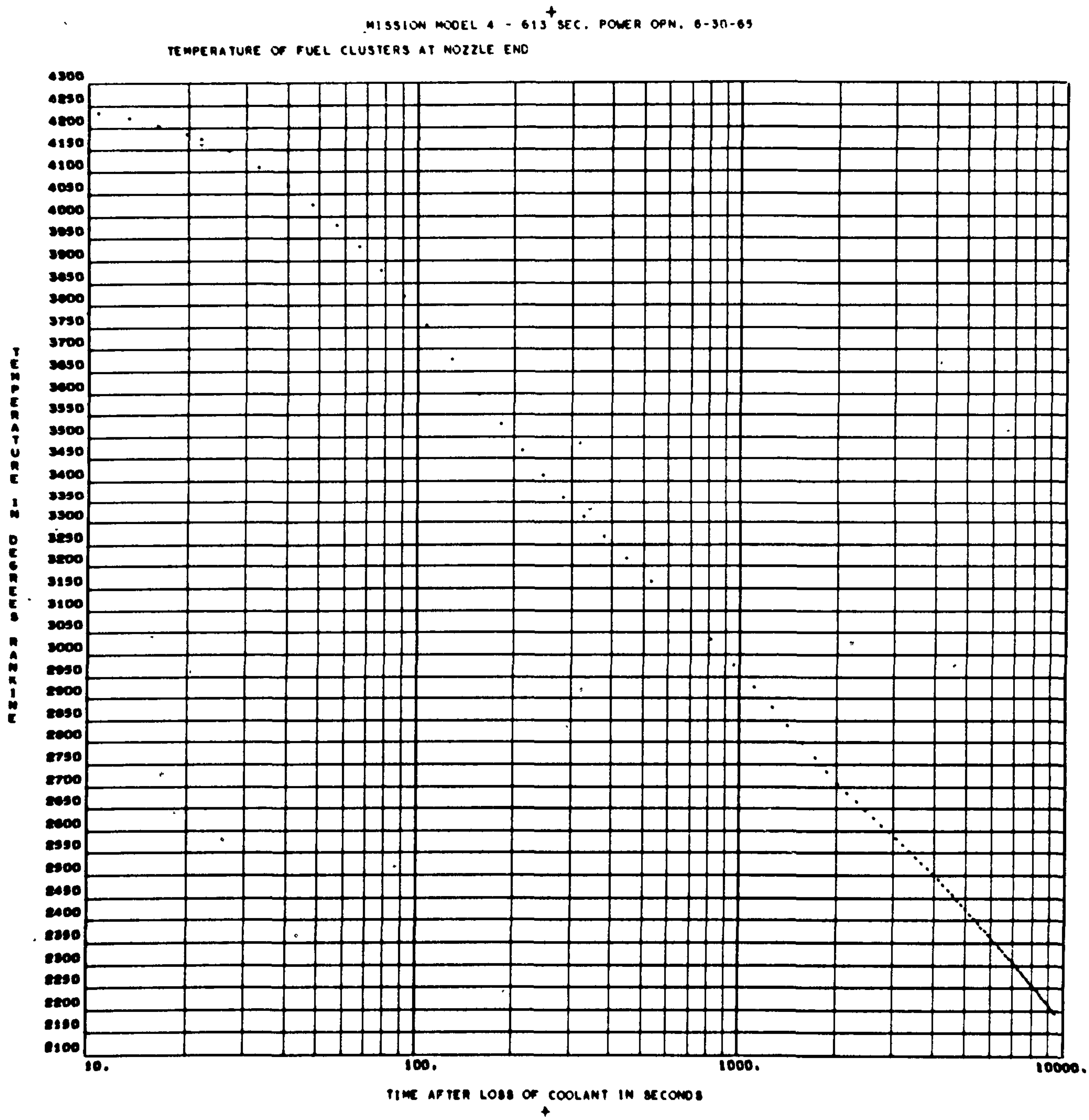

FIGURE 2.16

613 SECOND REACTOR OPERATION

TEMPERATURE OF FUEL CLUSTERS AT NOZZLE END

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TEMPERATURE OF GRAPHITE REFLECTOR AT NOZZLE END

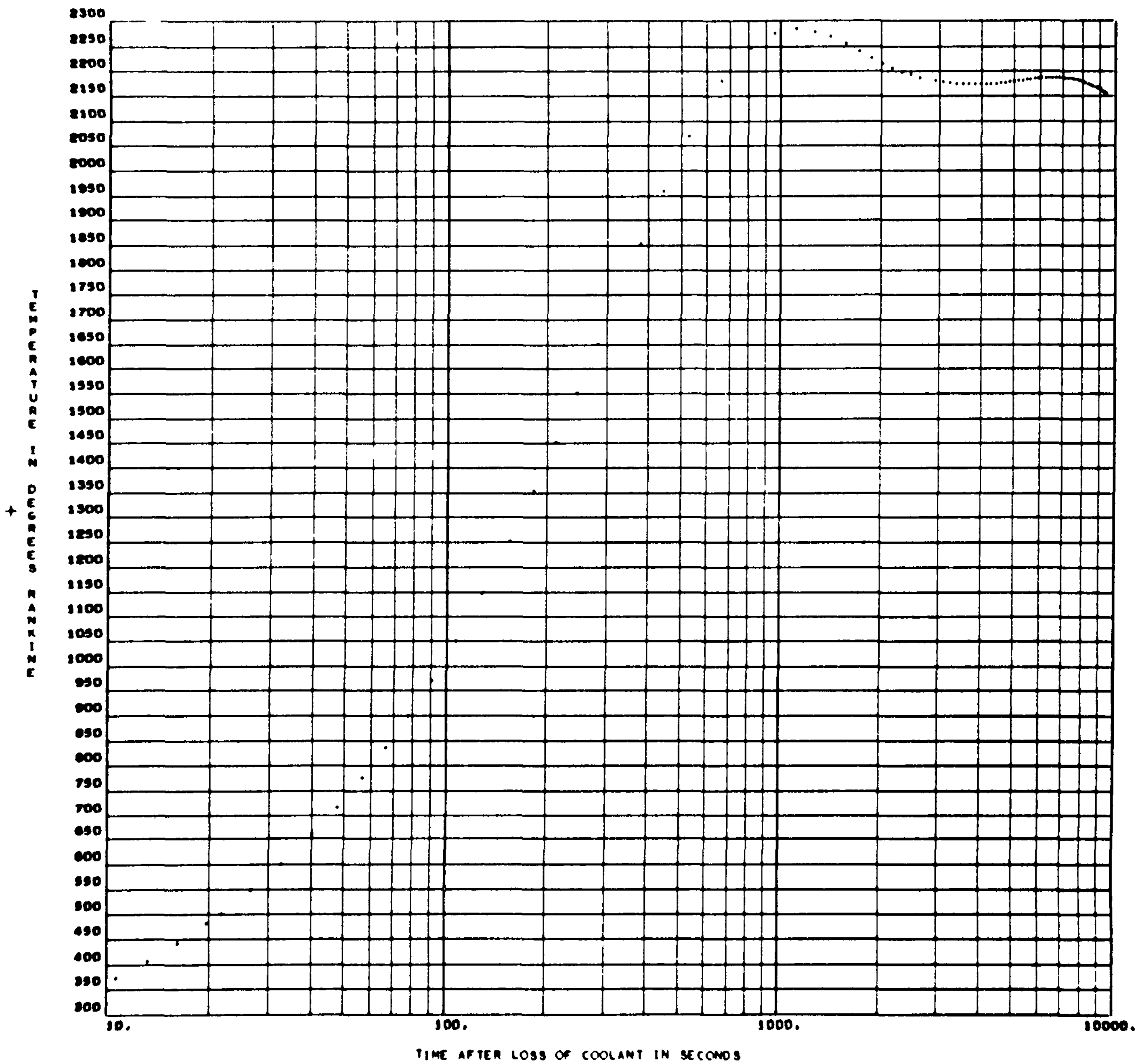

FIGURE 2.17

613 SECOND REACTOR OPERATION

TEMPERATURE OF GRAPHITE REFLECTOR AT NOZZLE END

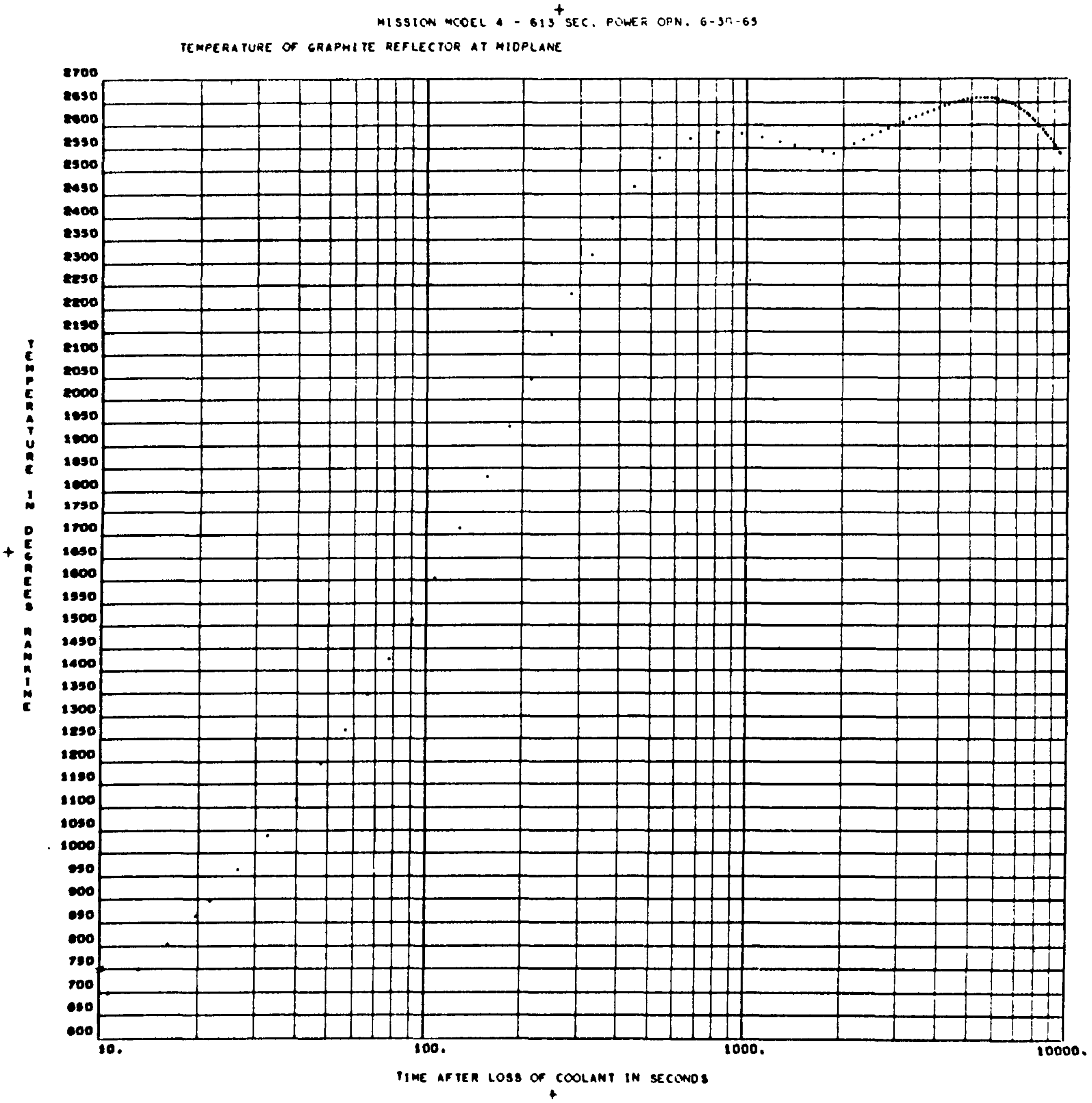

FIGURE 2.18

613 SECOND REACTOR OPERATION

TEMPERATURE OF GRAPHITE REFLECTOR AT CORE MIDPLANE

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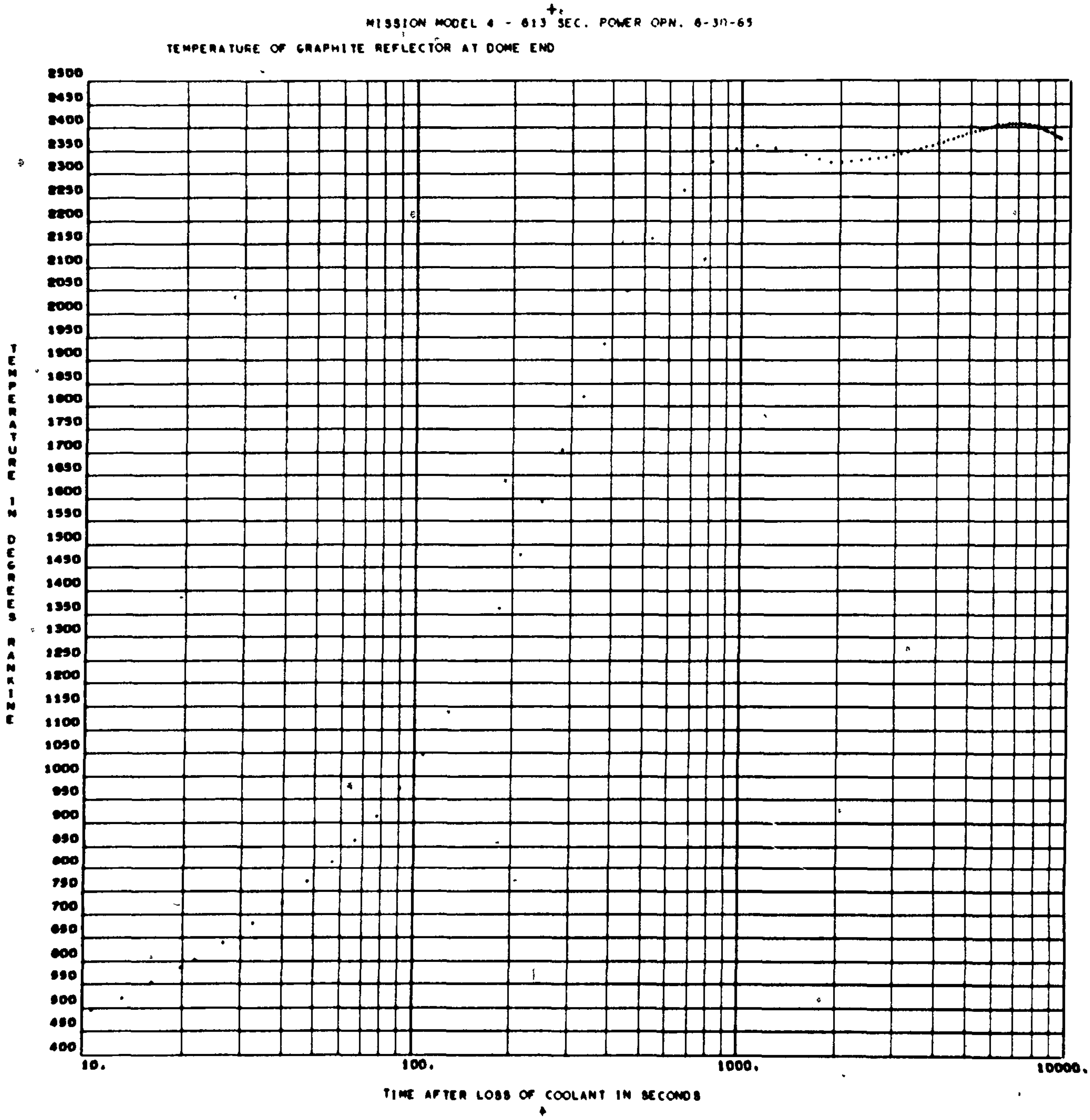

FIGURE 2.19

613 SECOND REACTOR OPERATION

TEMPERATURE OF GRAPHITE REFLECTOR AT DOME END

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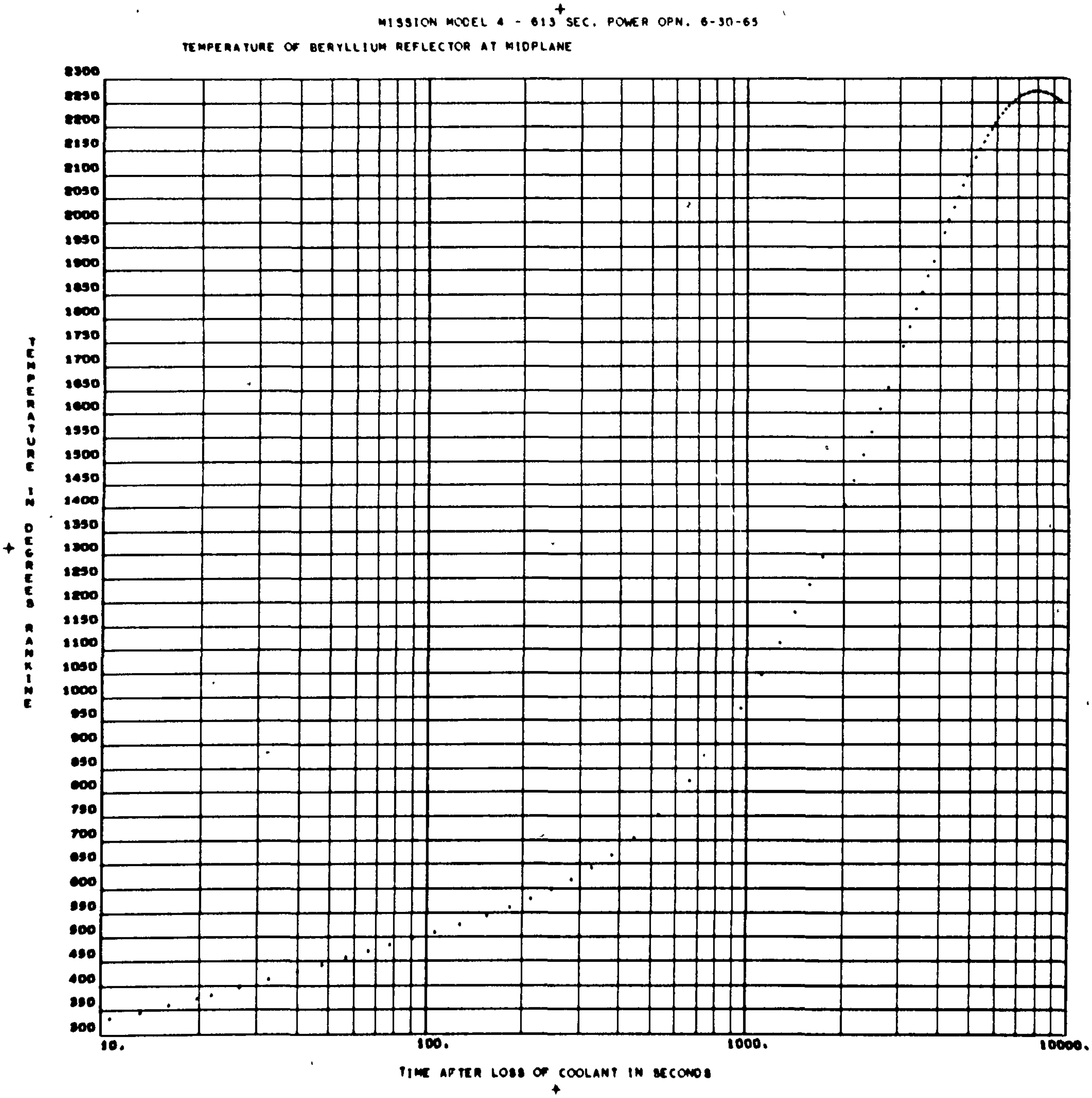

FIGURE 2.20

613 SECOND REACTOR OPERATION

TEMPERATURE OF BERYLLIUM REFLECTOR AT CORE MIDPLANE

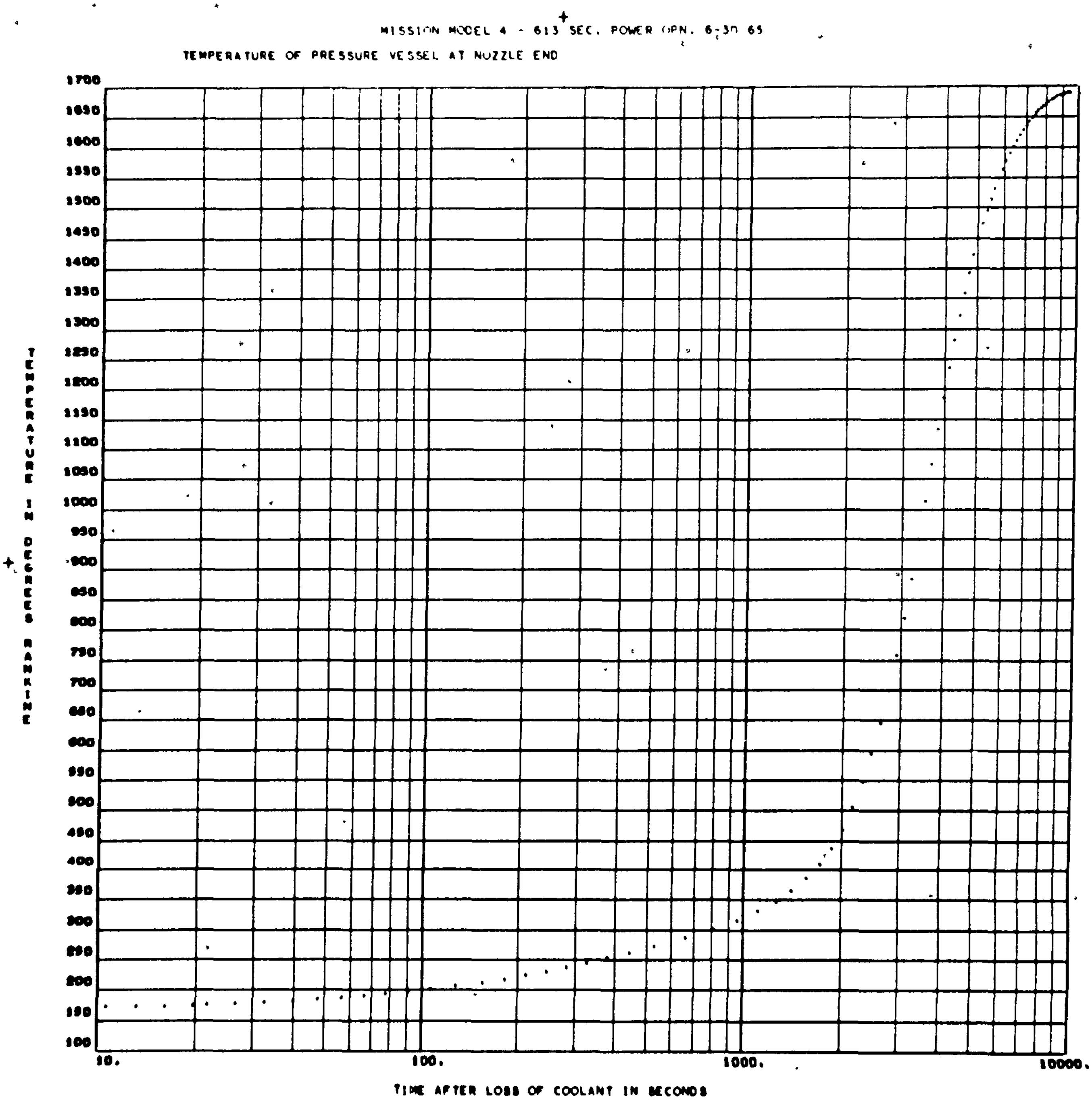

FIGURE 2.21

613 SECOND REACTOR OPERATION

TEMPERATURE OF PRESSURE VESSEL AT NOZZLE END

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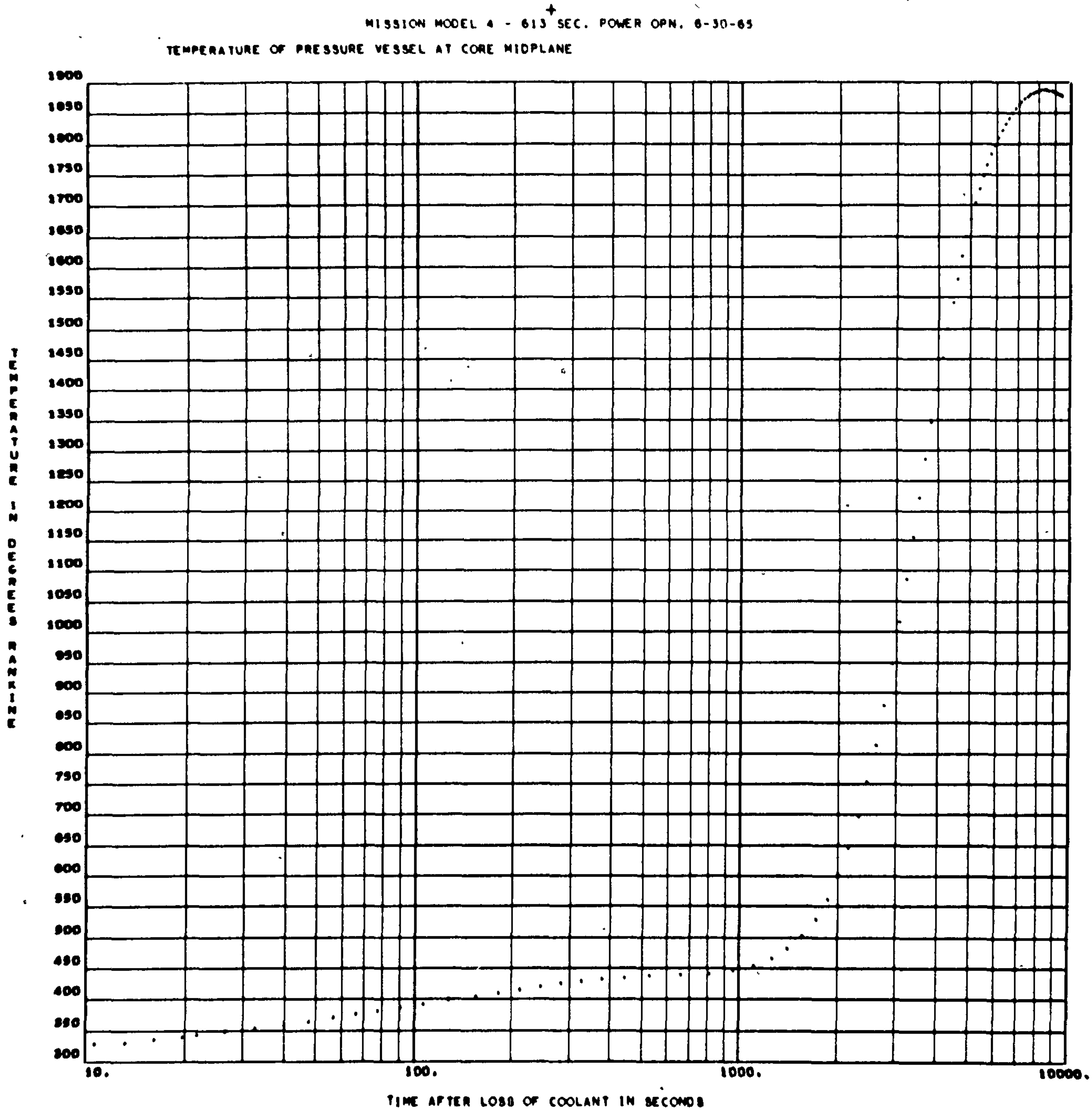

FIGURE 2.22

613 SECOND REACTOR OPERATION

TEMPERATURE OF PRESSURE VESSEL AT CORE MIDPLANE

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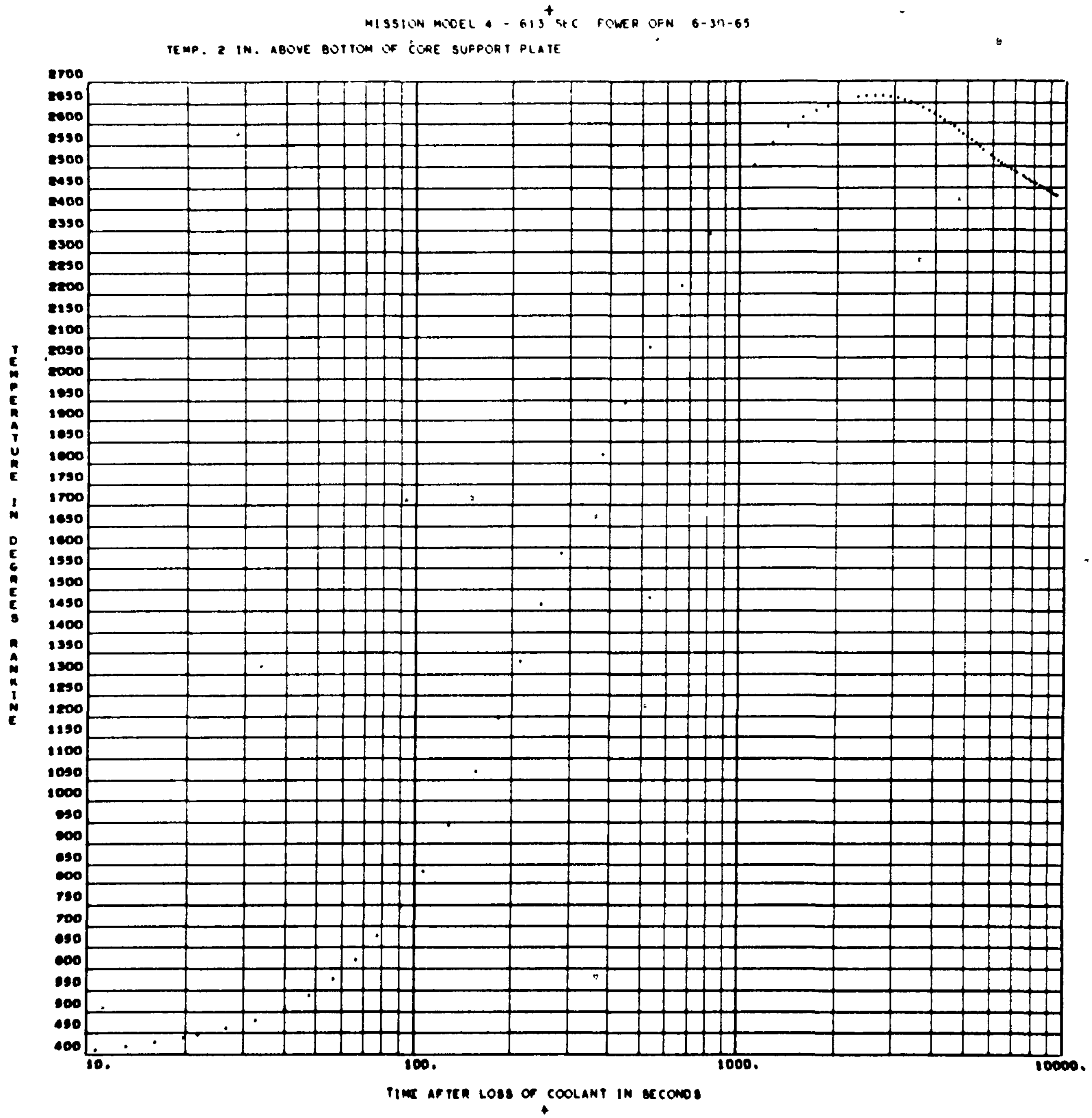

FIGURE 2.23

613 SECOND REACTOR OPERATION

TEMPERATURE OF CENTER OF CORE SUPPORT PLATE

2.2.1.1.3 TEMPERATURE DISTRIBUTIONS AND FISSION PRODUCT INVENTORIES FOR ORBITAL START CASES

Section 2.2.1.1.3 contains a series of curves and tables which present a comparison of reactor component temperatures of interest and fission product inventory data for the orbital start cases.

It was noted earlier that disassembly of the core might be presumed to begin when either of the following occurs:

1.) The lateral support system fails or,

2.) The pressure vessel melts.

Reference to Figures 2.17, 2.18, and 2.19 shows that the graphite reflector temperature at the core exit rises somewhat more slowly than at any other location in the reflector. Since the lateral support system is in intimate contact with the inner reflector and can be presumed to be at the reflector temperature, the time required for complete deterioration of the support system can be predicted on the basis of the temperature at the core exit end of the reflector. The range of temperatures at this location are displayed in Figure 2.24 for the orbital start cases considered.

Reference to Figure 2.10 indicates that the hottest section of the pressure vessel is at the core midplane. Thus, initial pressure vessel melting will occur at this location. The temperature history of this reactor location is given in Figure 2.25 for the four orbital cases. In both Figures 2.24 and 2.25, the time interval over which component failures occurs is indicated.

Tables 2.2 and 2.3 present data on the reduction of fission product inventory due to diffusion at each of the failure times derived from Figures 2.24 and 2.25.

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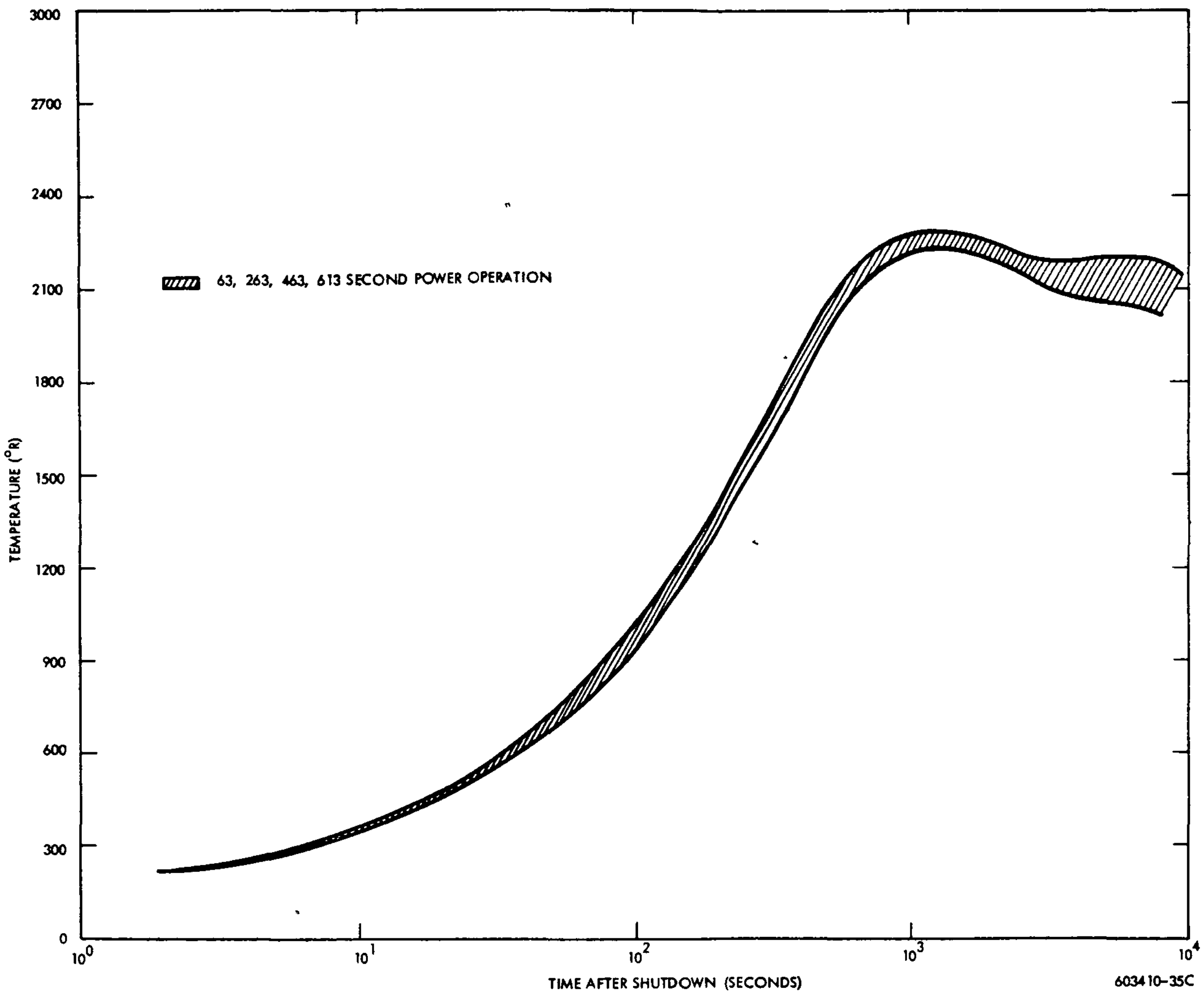

FIGURE 2.24

TEMPERATURE OF GRAPHITE REFLECTOR AT CORE EXIT

VS TIME AFTER FAILURE

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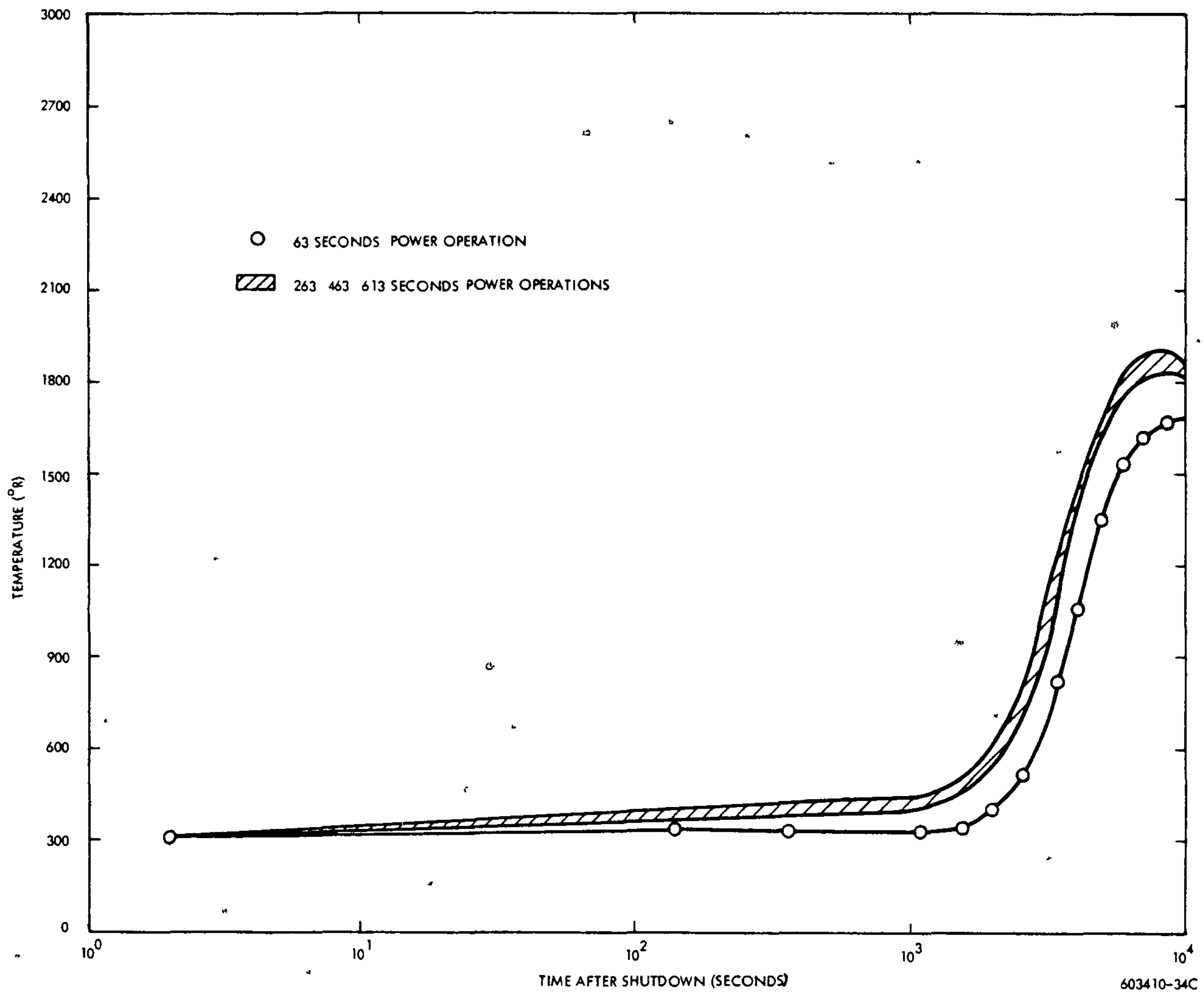

FIGURE 2.25

TEMPERATURE OF PRESSURE VESSEL NEAR CORE MIDPLANE

VS TIME AFTER FAILURE

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TABLE 2.2

FISSION PRODUCT INVENTORY AT SELECTED TIMES FOLLOWING FAILURE OF THE NUCLEAR STAGE DURING INITIAL FIRING

Failure Time (sec)	Inventory on Failure of Lateral Support System (microcuries/fuel element)			Inventory on Pressure Vessel Melting (microcuries/fuel element)		
	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction
63	1.89×10^{10}	4.41×10^{10}	57.2	1.74×10^{8}	7.09×10^{8}	74.2
263	1.43×10^{11}	2.46×10^{11}	41.8	1.95×10^{9}	8.56×10^{9}	77.3
463	2.21×10^{11}	3.69×10^{11}	40.0	5.16×10^{9}	2.09×10^{10}	75.2
613	2.66×10^{11}	4.40×10^{11}	39.4	5.13×10^{9}	2.26×10^{10}	77.0

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TABLE 2.3

SOURCE TERM ON THE EARTH'S SURFACE DUE TO RE-ENTRY OF A SINGLE FUEL ELEMENT FOLLOWING FAILURE OF THE NUCLEAR STAGE DURING INITIAL FIRING

Case 1 - Fuel Element Released when Lateral Support System Fails

Engine Failure Time (sec)	Fission Product Activity (microcuries/fuel element)			Gamma Source Strength (MEV/sec-fuel element)			Beta Source Strength (MEV/sec-fuel element)		
	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction
63	1.72×10^{7}	2.47×10^{7}	30.4	2.94×10^{11}	4.57×10^{11}	35.7	1.80×10^{11}	2.83×10^{11}	36.4
263	3.17×10^{6}	4.89×10^{6}	35.2	5.29×10^{10}	7.02×10^{10}	24.6	2.42×10^{10}	4.81×10^{10}	49.7
463	9.86×10^{5}	1.48×10^{6}	33.3	1.32×10^{10}	1.65×10^{10}	20.0	9.97×10^{9}	1.71×10^{10}	41.7
613	2.56×10^{5}	3.87×10^{5}	33.9	5.88×10^{8}	1.06×10^{9}	44.5	4.15×10^{9}	6.11×10^{9}	32.1

Case 2 - Fuel Element Released when Pressure Vessel Melts

Engine Failure Time (sec)	FP With Diffusion	FP Without Diffusion	% Reduction	Gamma With Diffusion	Gamma Without Diffusion	% Reduction	Beta With Diffusion	Beta Without Diffusion	% Reduction
63	1.12×10^{7}	2.33×10^{7}	51.9	1.61×10^{11}	4.31×10^{11}	62.6	1.11×10^{11}	2.64×10^{11}	58.0
263	1.87×10^{6}	4.89×10^{6}	61.8	3.63×10^{10}	7.02×10^{10}	48.3	9.48×10^{9}	4.81×10^{10}	80.3
463	5.48×10^{5}	1.48×10^{6}	63.0	1.07×10^{10}	1.65×10^{10}	35.2	3.03×10^{9}	1.71×10^{10}	82.3
613	9.03×10^{4}	3.87×10^{5}	76.6	3.17×10^{8}	1.06×10^{9}	70.1	1.04×10^{9}	6.11×10^{9}	83.0

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2.2.1.2 SUB-ORBITAL START CASES

The temperature profiles prior to core disassembly, which would result from failure during a sub-orbital mission, would be the same as for the corresponding orbital case. However, due to the more rapid re-entry the fission product inventory which would exist in the reactor core at the time of impact would be much greater. Two possible times for disassembly have been considered in these analyses. One of these is at the time of failure of the lateral support system. Another is when the reactor reaches 80,000 feet, at which time the maximum re-entry forces on the pressure vessel exist.[6] Table 2.4 is a tabulation of fission product inventory, with and without diffusion, at both possible disassembly times for each of three failure times. Table 2.5 is a compilation of fission product activity, gamma souce strength, and beta source strength vs failure time for both possible disassembly times.

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TABLE 2.4

FISSION PRODUCT INVENTORY AT SELECTED TIMES FOLLOWING FAILURE

OF THE NUCLEAR STAGE SUBSEQUENT TO SUBORBITAL START

Failure Time (sec)	Inventory on Failure of Lateral Support System (microcuries/fuel element)			Inventory when Reactor Reaches 80,000 feet (microcuries/fuel element)		
	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction
63	1.89×10^{10}	4.41×10^{10}	57.2	6.49×10^{9}	1.29×10^{10}	49.7
263	1.43×10^{11}	2.46×10^{11}	41.8	4.51×10^{10}	8.74×10^{10}	48.4
463	2.21×10^{11}	3.69×10^{11}	40.0	5.72×10^{10}	1.16×10^{11}	50.8

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TABLE 2.5

SOURCE TERM ON THE EARTH'S SURFACE DUE TO RE-ENTRY
OF A SINGLE FUEL ELEMENT FOLLOWING
FAILURE OF THE NUCLEAR STAGE SUBSEQUENT TO SUBORBITAL START

Case 1 - Fuel Element Released when Lateral Support System Fails

Engine Failure Time (sec)	Fission Product Activity (microcuries/fuel element)			Gamma Source Strength (MEV/sec-fuel element)			Beta Source Strength (MEV/sec-fuel element)		
	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction
63	3.78×10^{9}	6.73×10^{9}	44.1	1.71×10^{14}	3.08×10^{14}	44.5	7.85×10^{13}	1.59×10^{14}	50.6
263	3.14×10^{10}	5.21×10^{10}	39.7	1.42×10^{15}	2.36×10^{15}	39.8	6.62×10^{14}	1.22×10^{15}	45.7
463	4.93×10^{10}	8.04×10^{10}	38.7	2.13×10^{15}	3.54×10^{15}	39.8	1.04×10^{15}	1.87×10^{15}	44.4
Case 2 - Fuel Element Released when Intact Reactor Reaches 80,000 feet									
63	4.05×10^{9}	8.14×10^{9}	50.2	1.93×10^{14}	3.83×10^{14}	49.6	8.41×10^{13}	1.95×10^{14}	56.9
263	3.05×10^{10}	5.97×10^{10}	48.9	1.42×10^{15}	2.76×10^{15}	48.6	6.32×10^{14}	1.41×10^{15}	55.2
463	4.26×10^{10}	8.78×10^{10}	51.5	1.86×10^{15}	3.91×10^{15}	52.4	8.73×10^{14}	2.05×10^{15}	57.4

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2.2.1.3 RE-START CASES

Temperature distributions and histories for all components closely approximate those from the initial firing. This can be seen in the case of the graphite reflector (nozzle end) and pressure vessel (core midplane) in Figures 2.26 and 2.27.

Four periods of reactor operation have been considered for the second firing in the restart cases. These are: 63 seconds, 245 seconds, 463 seconds, and 613 seconds. The temperature of the graphite reflector at core exit and of the pressure vessel at core midplane following failure at these times are given in Figures 2.26 and 2.27 respectively.[6]

Source term data for these cases are given in Tables 2.6 and 2.7.

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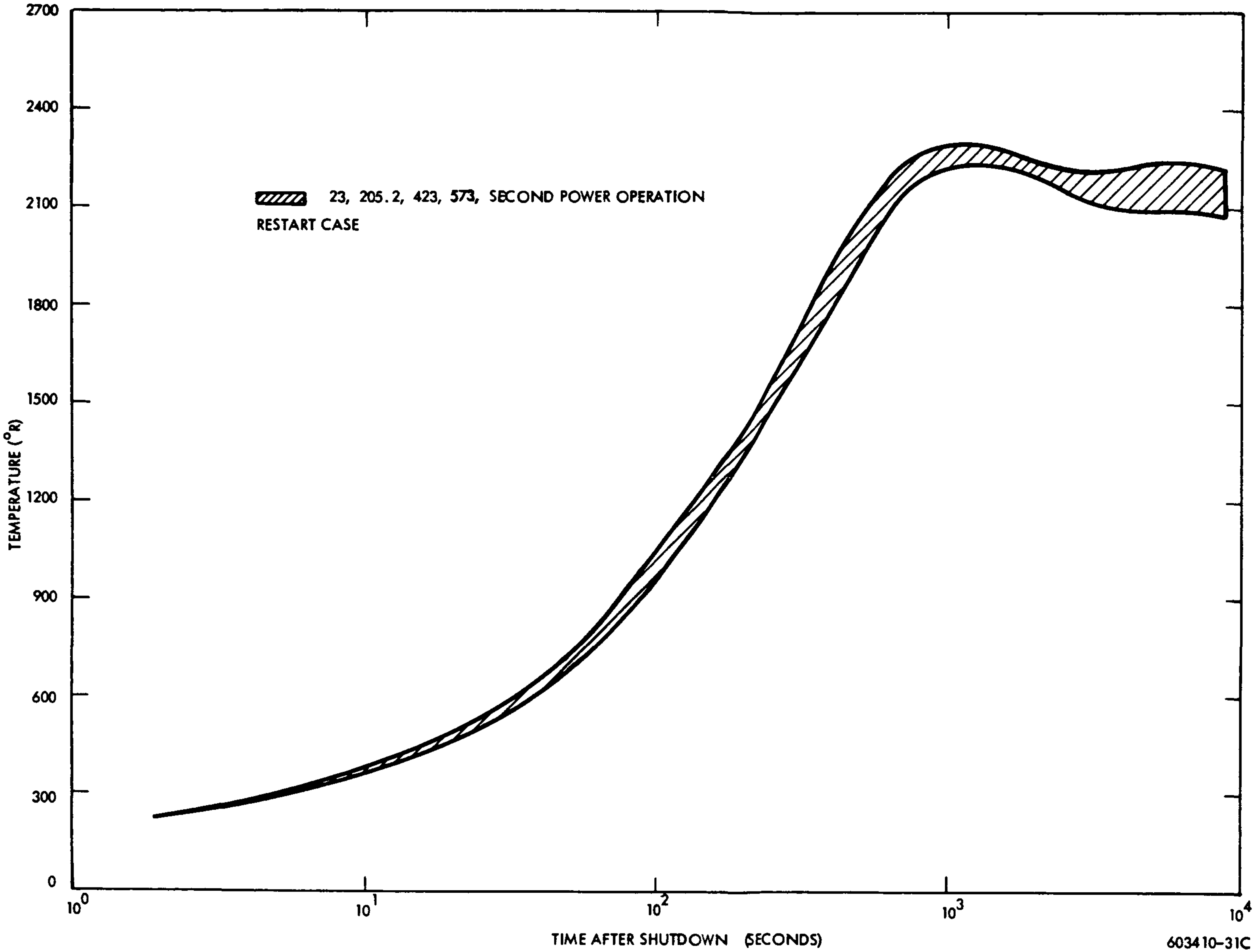

TEMPERATURE OF GRAPHITE REFLECTOR AT CORE EXIT
VS TIME AFTER FAILURE

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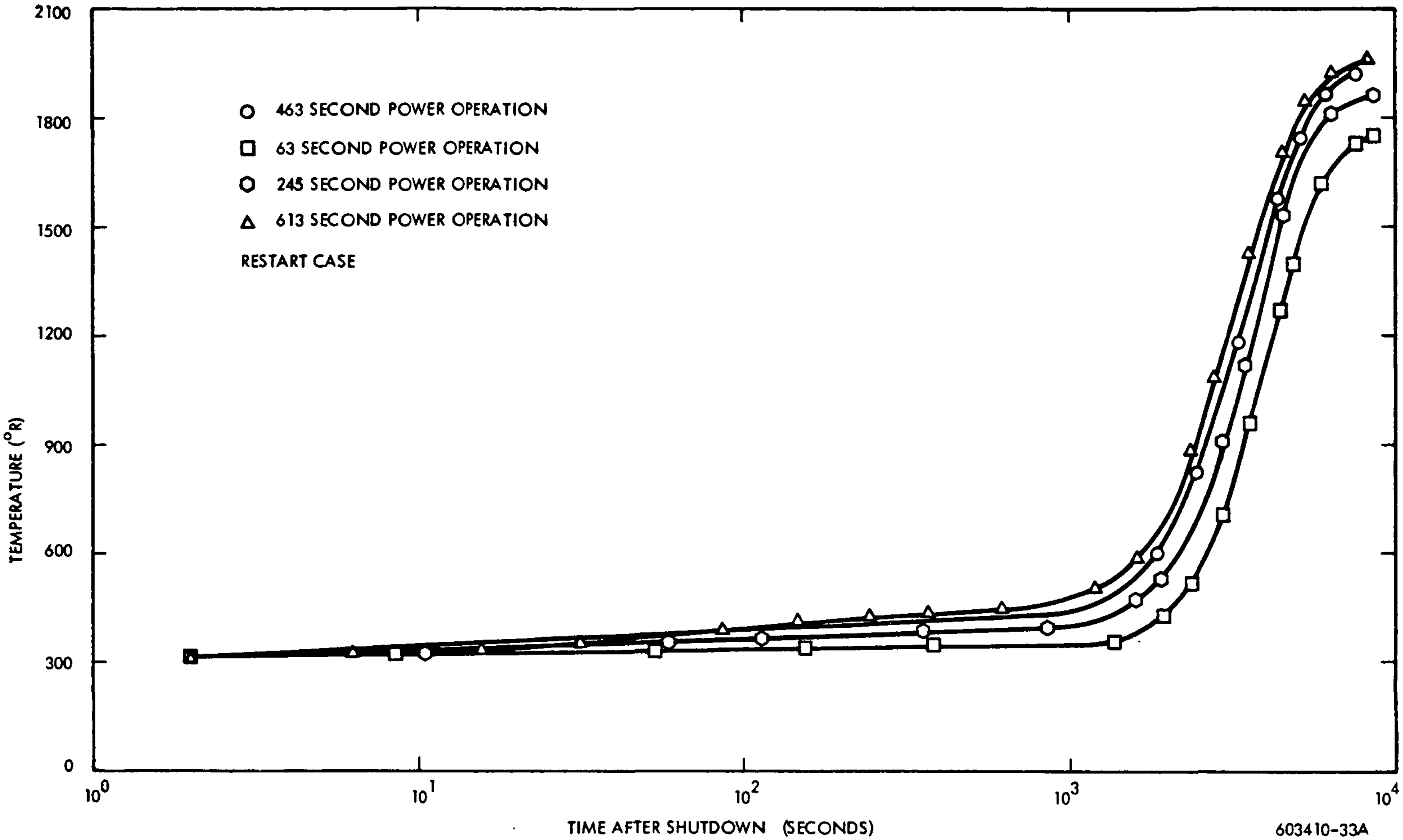

FIGURE 2.27

TEMPERATURE OF PRESSURE VESSEL NEAR CORE MIDPLANE

VS TIME AFTER FAILURE

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TABLE 2.6

FISSION PRODUCT INVENTORY AT SELECTED TIMES FOLLOWING
RE-START AND FAILURE OF THE NUCLEAR STAGE

Failure Time (sec)	Inventory on Failure of Lateral Support System (microcuries/fuel element)			Inventory on Pressure Vessel Melting (microcuries/fuel element)		
	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction
63	5.61×10^{10}	1.08×10^{11}	48.1	3.09×10^{9}	1.25×10^{10}	75.1
245	1.64×10^{11}	2.82×10^{11}	41.8	4.72×10^{9}	2.08×10^{10}	77.4
463	2.57×10^{11}	4.25×10^{11}	39.5	7.71×10^{9}	3.34×10^{10}	76.9
613	2.81×10^{11}	4.69×10^{11}	40.0	9.51×10^{9}	4.13×10^{10}	77.0

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TABLE 2.7

SOURCE TERM ON THE EARTH'S SURFACE DUE TO RE-ENTRY
OF A SINGLE FUEL ELEMENT FOLLOWING
RESTART AND FAILURE OF THE NUCLEAR STAGE

Case 1 - Fuel Element Released when Lateral Support System Fails

Engine Failure Time After Restart (sec)	Fission Product Activity (microcuries/fuel element)			Gamma Source Strength (MEV/sec-fuel element)			Beta Source Strength (MEV/sec-fuel element)		
	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction	With Diffusion	Without Diffusion	% Reduction
63	5.30×10^{7}	7.57×10^{7}	30.0	1.20×10^{12}	1.53×10^{12}	21.6	4.80×10^{11}	7.22×10^{11}	33.5
245	1.30×10^{7}	1.96×10^{7}	33.7	2.44×10^{11}	2.91×10^{11}	16.2	9.72×10^{10}	1.94×10^{11}	49.9
463	2.54×10^{6}	3.64×10^{6}	30.2	3.60×10^{10}	4.07×10^{10}	11.5	2.46×10^{10}	4.23×10^{10}	41.8
613	5.28×10^{5}	8.07×10^{5}	34.6	1.27×10^{9}	2.20×10^{9}	42.3	8.45×10^{9}	1.28×10^{10}	34.0

Case 2 - Fuel Element Released when Pressure Vessel Melts

Engine Failure Time After Restart (sec)	Fission Product Activity With Diffusion	Fission Product Activity Without Diffusion	% Reduction	Gamma With Diffusion	Gamma Without Diffusion	% Reduction	Beta With Diffusion	Beta Without Diffusion	% Reduction
63	2.99×10^{7}	7.57×10^{7}	60.5	6.66×10^{11}	1.53×10^{12}	56.5	2.46×10^{11}	7.22×10^{11}	65.9
245	8.45×10^{6}	1.96×10^{7}	56.9	1.82×10^{11}	2.90×10^{11}	37.2	4.37×10^{10}	1.94×10^{11}	77.5
463	1.58×10^{6}	3.64×10^{6}	56.6	3.25×10^{10}	4.07×10^{10}	20.1	7.99×10^{9}	4.23×10^{10}	81.1
613	1.90×10^{5}	8.07×10^{5}	76.5	7.20×10^{8}	2.20×10^{9}	67.3	2.02×10^{9}	1.28×10^{10}	84.2

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III. CONCLUSIONS

The following conclusions may be drawn from the loss-of-coolant accidents which have been analyzed using the combined NOFLOW-FIPDIF program:

1.) The diffusion of fission products significantly lowers the reactor component temperatures for times greater than 1000 seconds after loss of coolant. Examples of the temperature reductions due to diffusion are: Core, 400°R; graphite reflector, 250°R; beryllium reflector, 350°R; pressure vessel, 350°R. These figures corresponds to a time of 4860 seconds following loss of coolant. The reactor had been operated for a period of 613 seconds prior to the accident.

2.) The reduction in the inventories of a number of nuclides, due to diffusion, has been determined for the 613 second operating period case. The ratio of the inventory with diffusion to that without diffusion has been determined at a time 5436 seconds after coolant loss. For the isotopes considered, this ratio varied from 1.0 (no significant loss) to .0002 (99.98 percent loss).

3.) For the three types of cases analyzed, orbital start, sub-orbital start, and orbital re-start, the reduction in total fission product activity at ground impact, due to diffusion, varied from 30% to 76%.

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[redacted]ergy Act - 1954

IV. REFERENCES

1. M. R. Trammell, "Analysis of the Effect of Post-Operational Heat on a NERVA Reactor", WANL-TME-957, October, 1964.

2. J. D. Cleary and G. T. Rymer, "Interim Report on Fission Product Diffusion Code (FIPDIF)", WANL-TME-958, September, 1964.

3. W. Knecht and W. L. Howarth, "NRX-A Reactor Model for Systems Analysis", WANL-TME-483, July 31, 1963.

4. Reactor Analysis, "Reactor Analysis of NRX-A, Volume III, Thermal and Fluid Flow Analysis", WANL-TNR-128, September, 1963.

5. G. T. Rymer, et al., "Interim Report on Fission Product Release from NERVA Fuel During Post Irradiation Thermal Anneals and During the NRX-A2 Test", WANL-TME-1165, July, 1965.

6. L. H. Grob,and M. R. Trammell, "Analysis of the Re-Entry Behavior of a NERVA Vehicle", WANL-TME-1174, June, 1965.

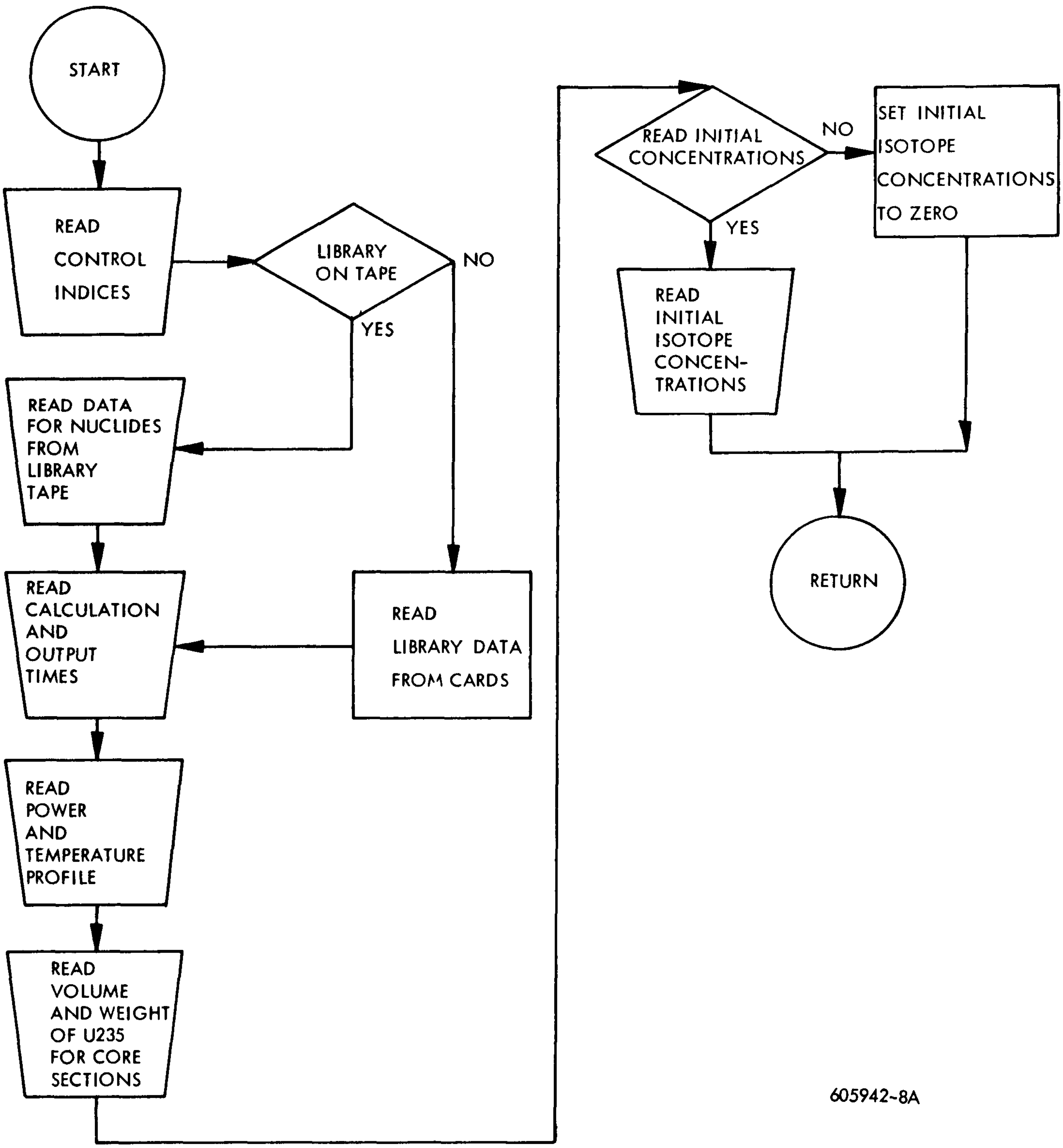
7. W. S. Brown, "WANL Source Term Program Status Report".

V. APPENDIX A

FLOW DIAGRAMS FOR COMPUTER PROGRAM

LINK I

SUBROUTINE FIPIN



605942-8A

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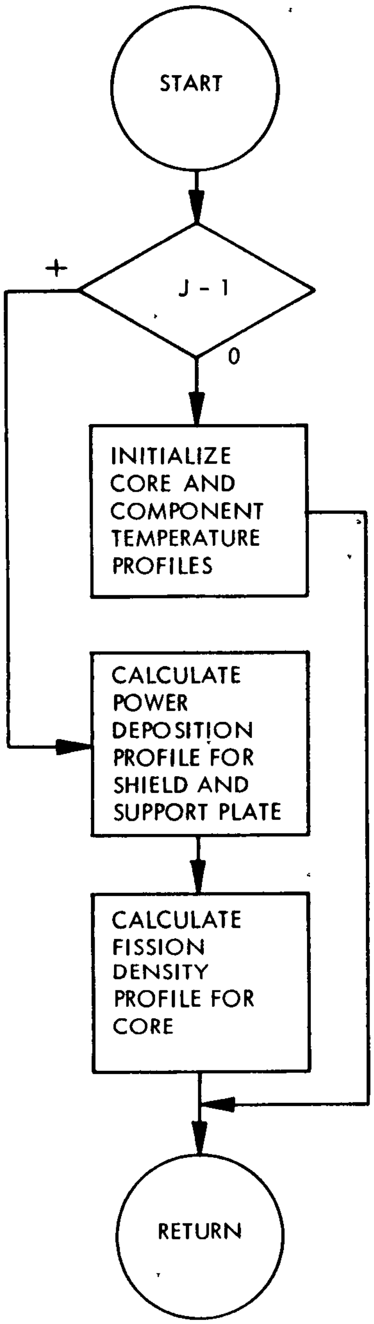

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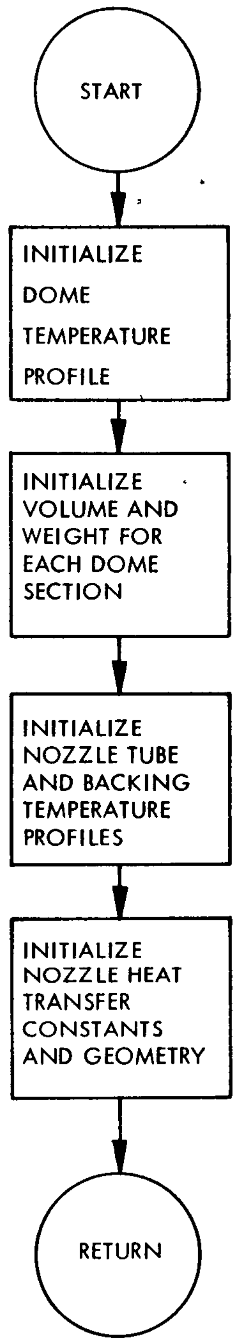

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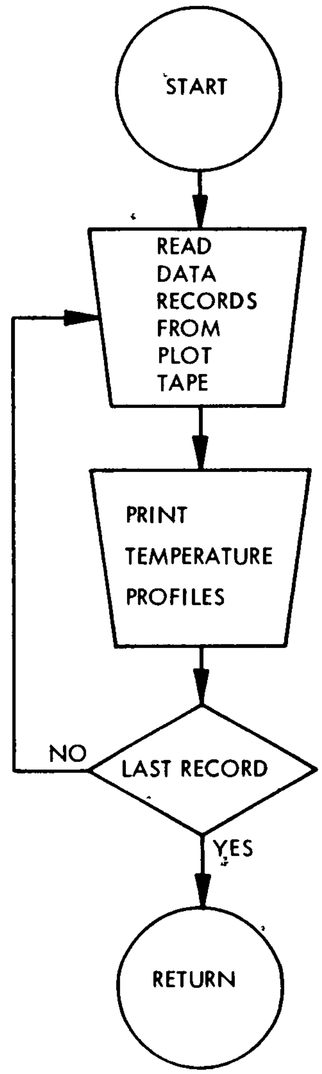

605942-7A

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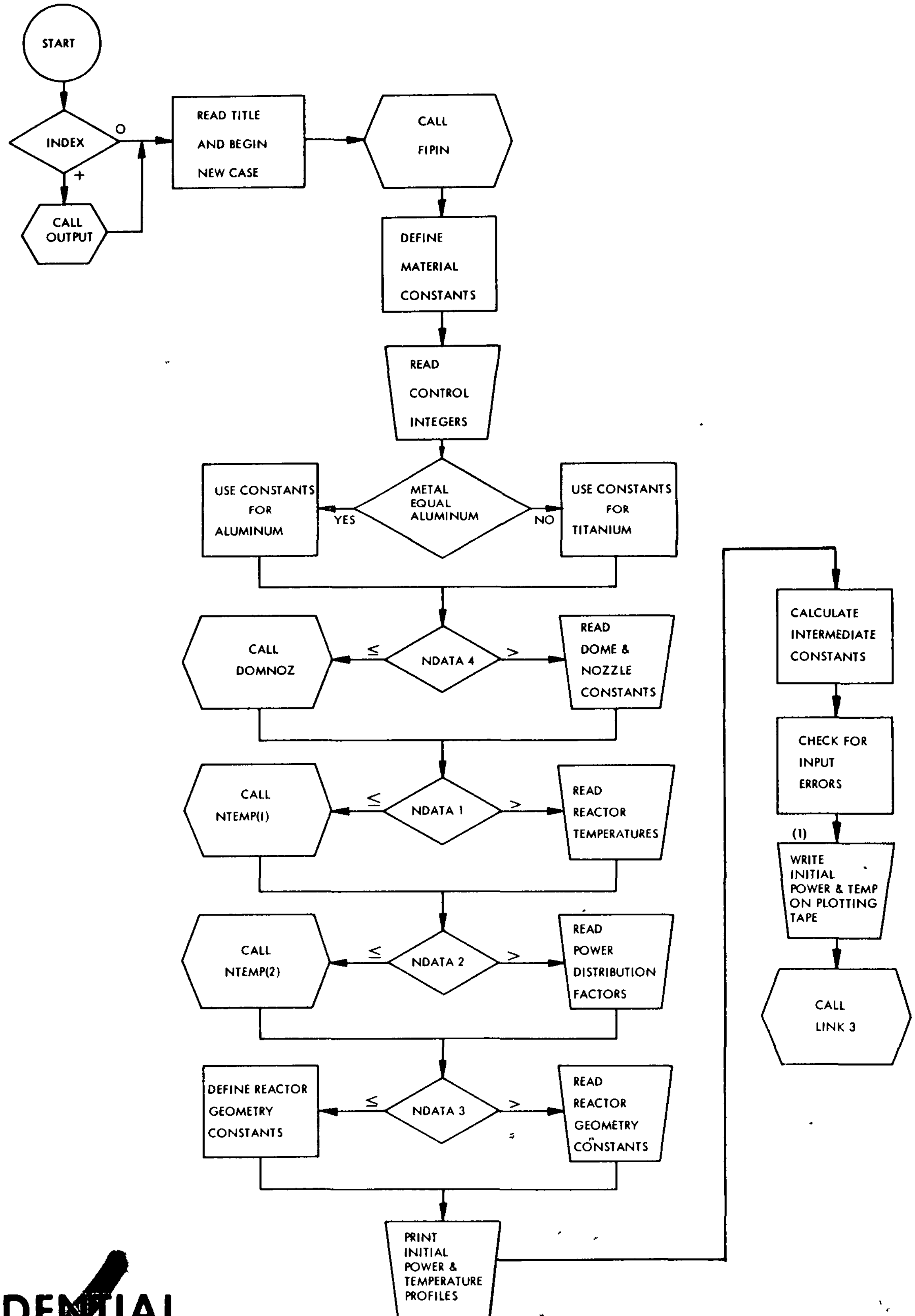

605942-11C

LINK II

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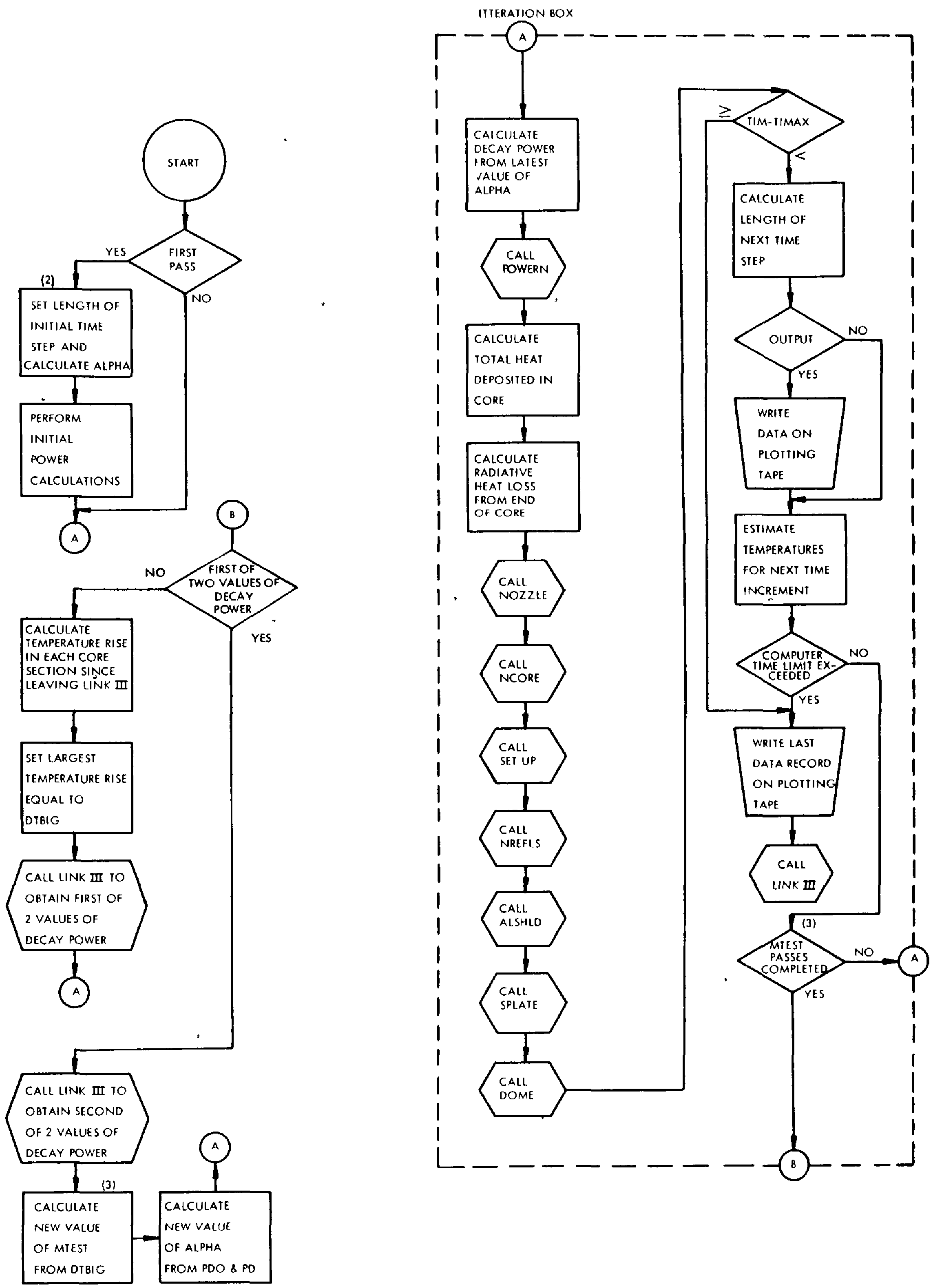
LINK 2 MAIN PROGRAM



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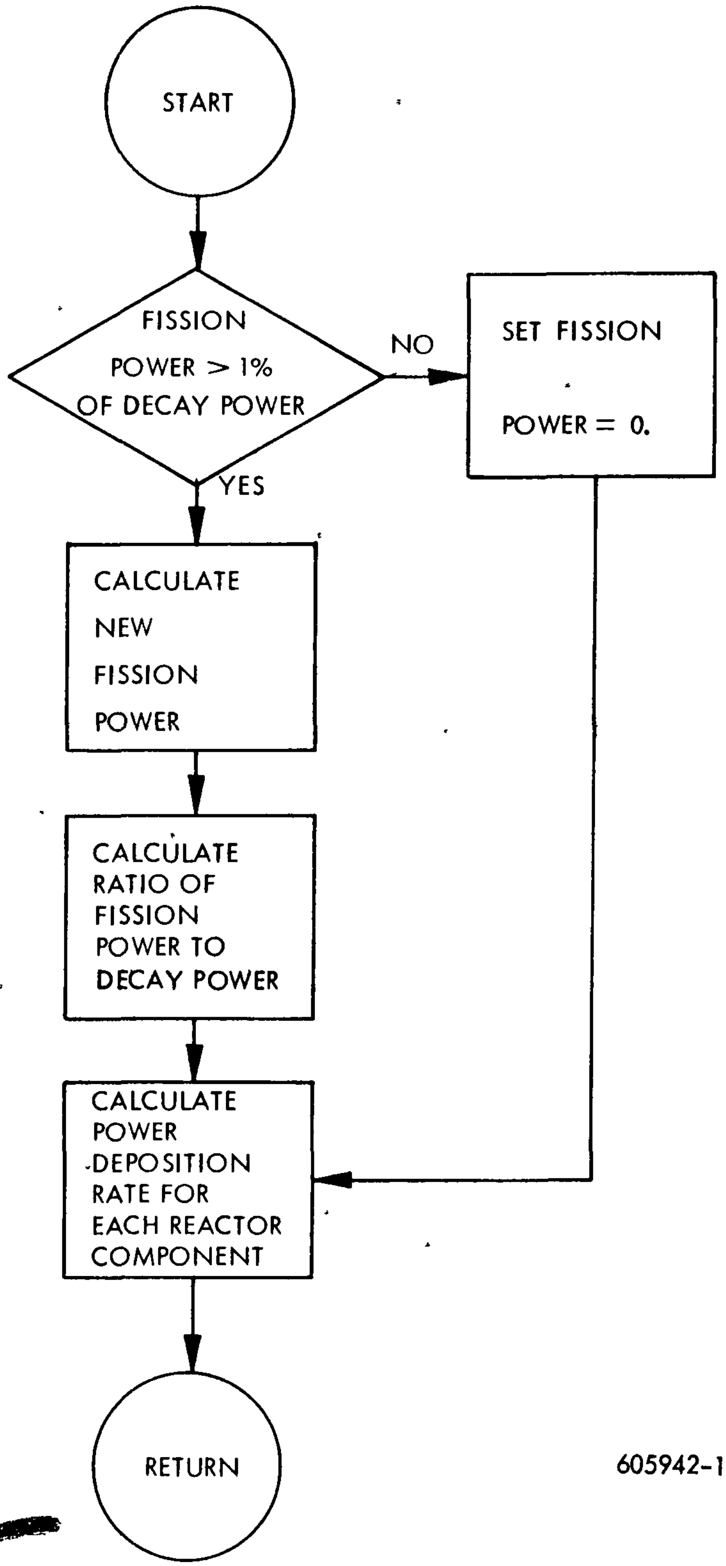
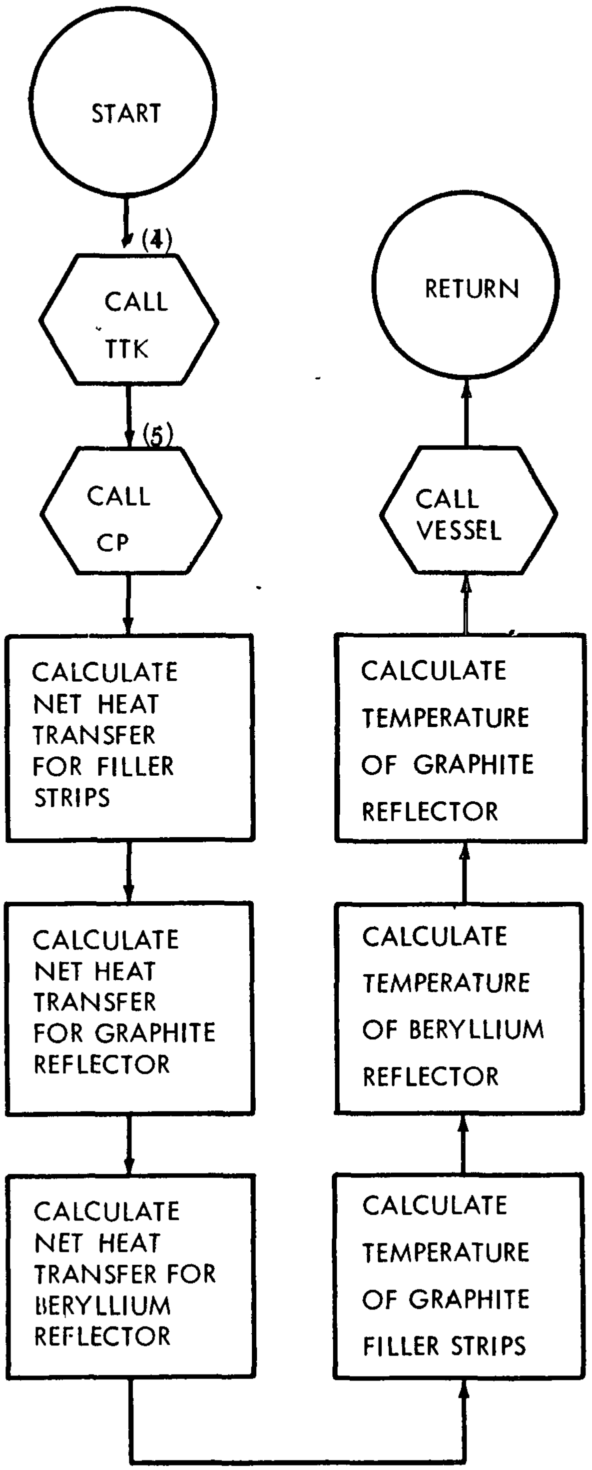
SUBROUTINE POWERN



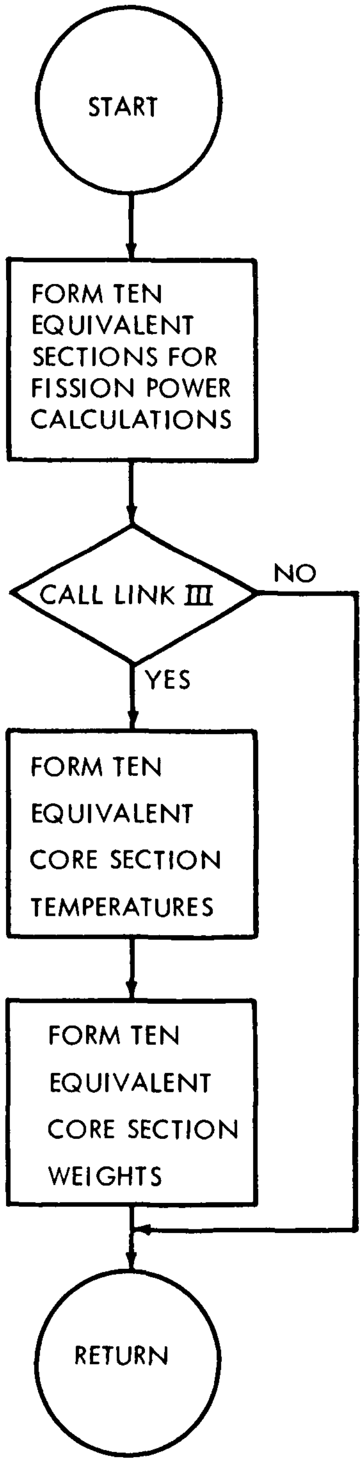
605942-14A

[redacted]

SUBROUTINE NREFLES



SUBROUTINE SETUP



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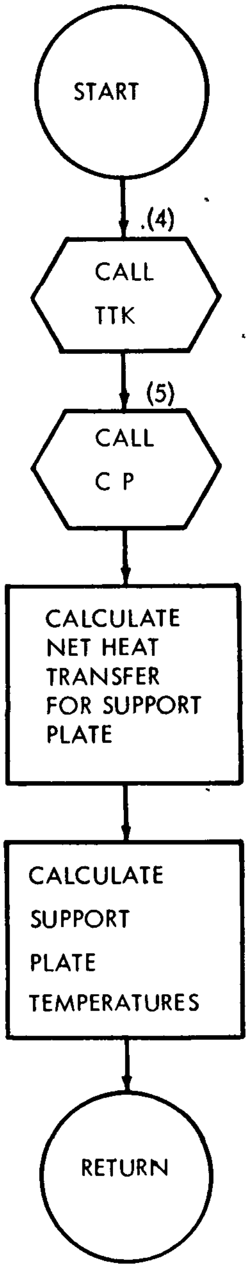
SUBROUTINE SPLATE



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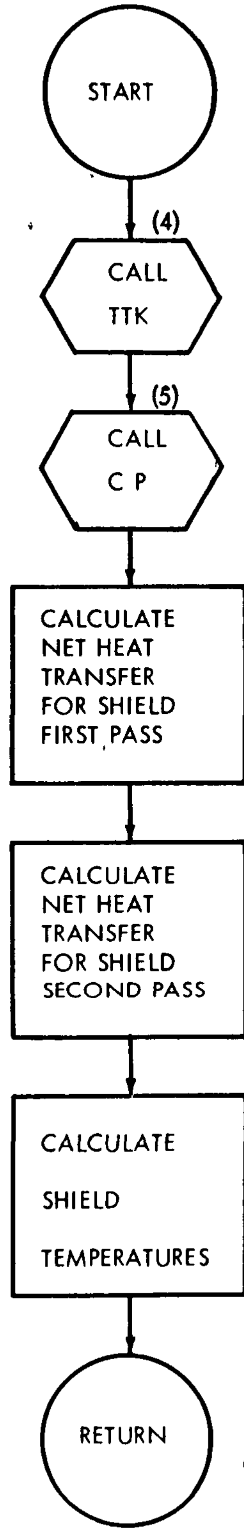

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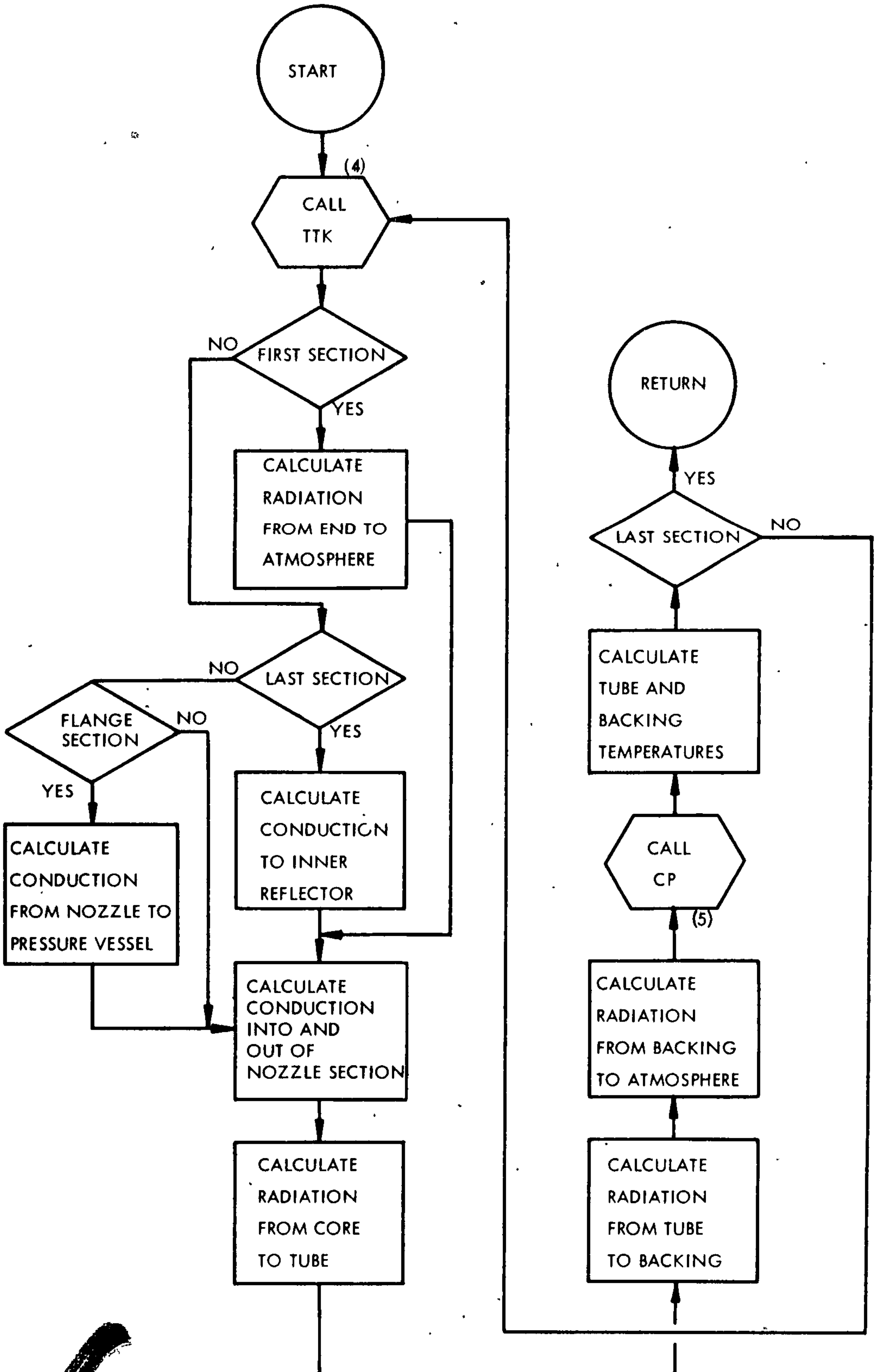
SUBROUTINE NOZZLE



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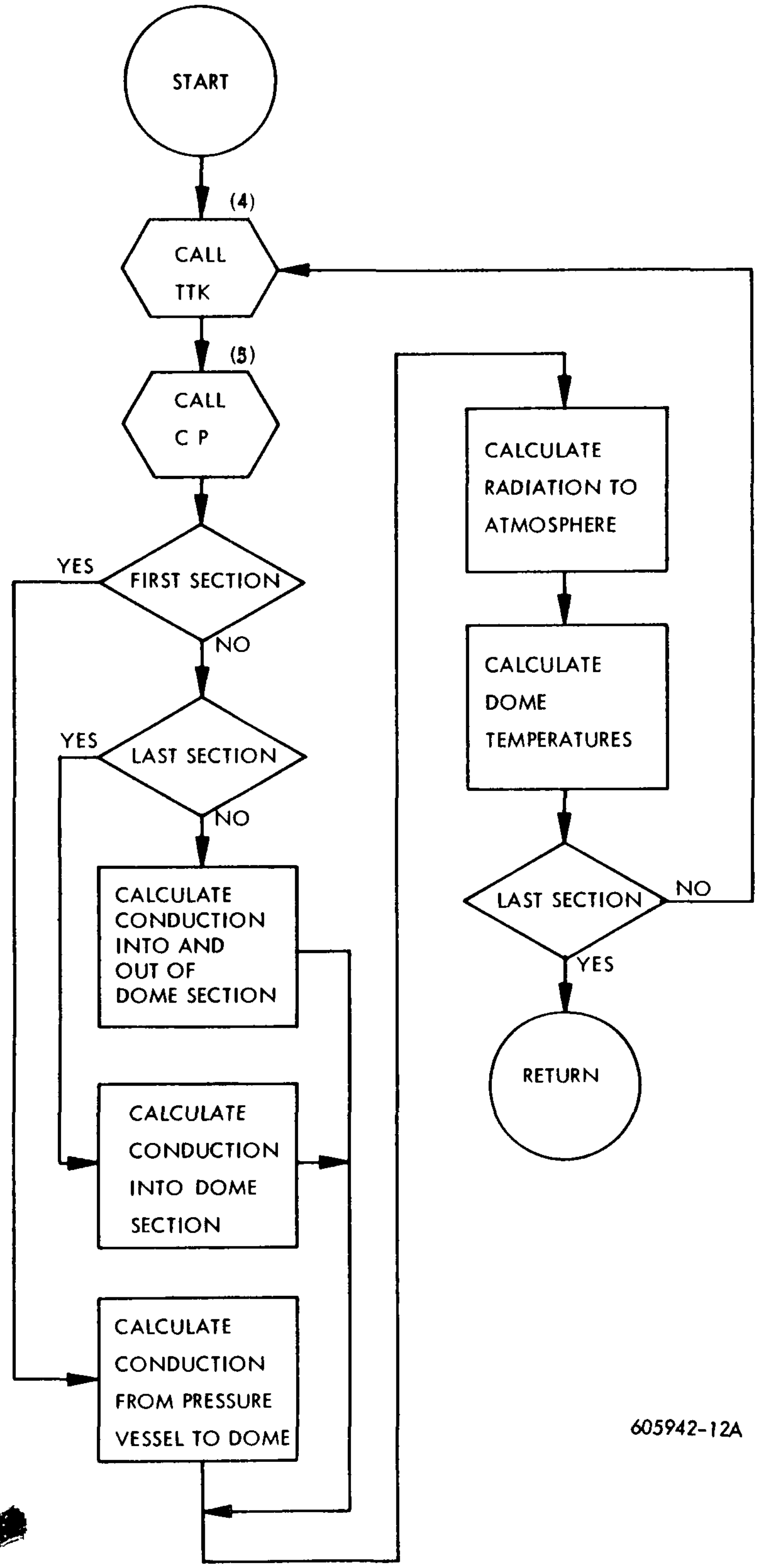
SUBROUTINE DOME



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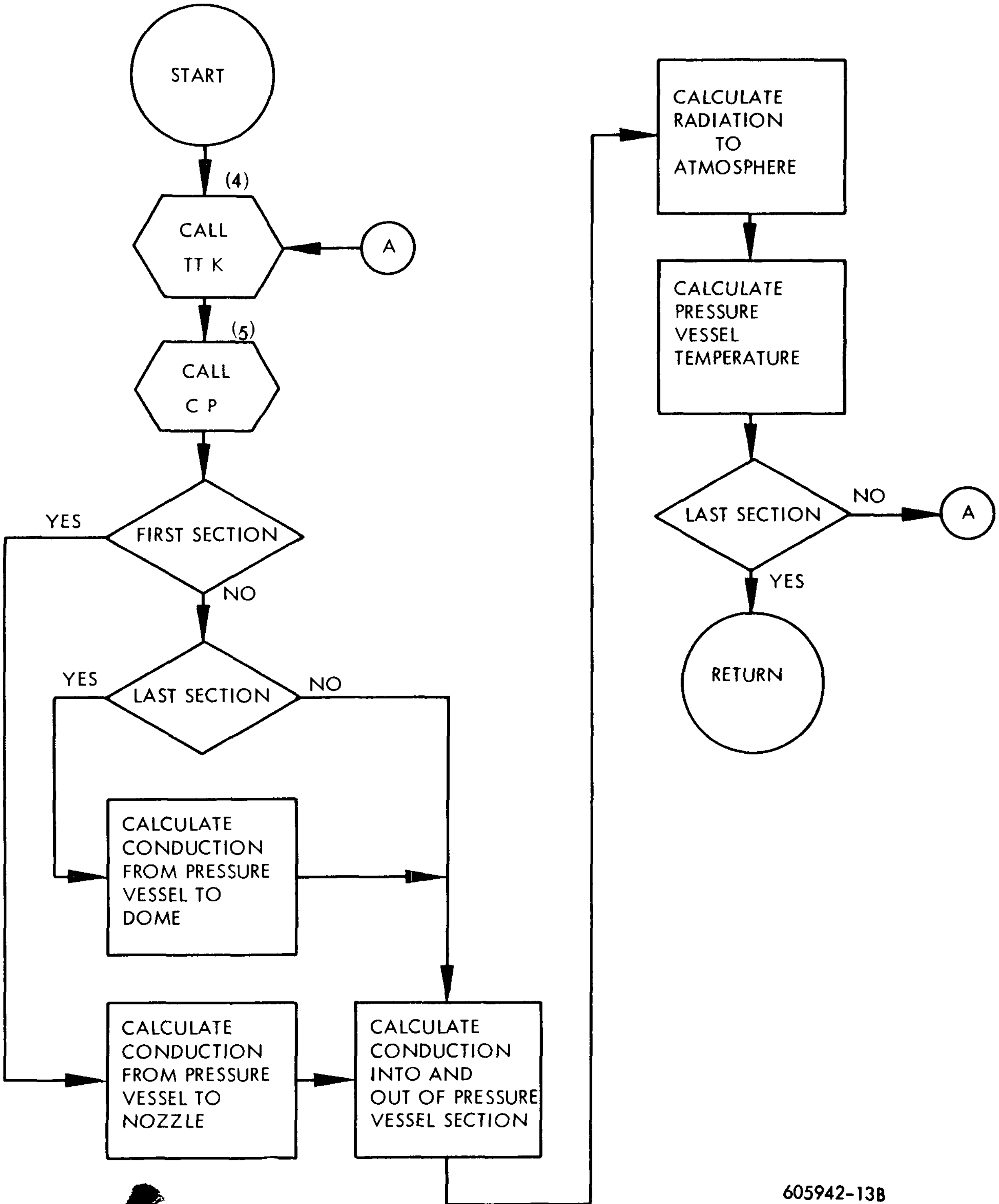
SUBROUTINE VESSEL



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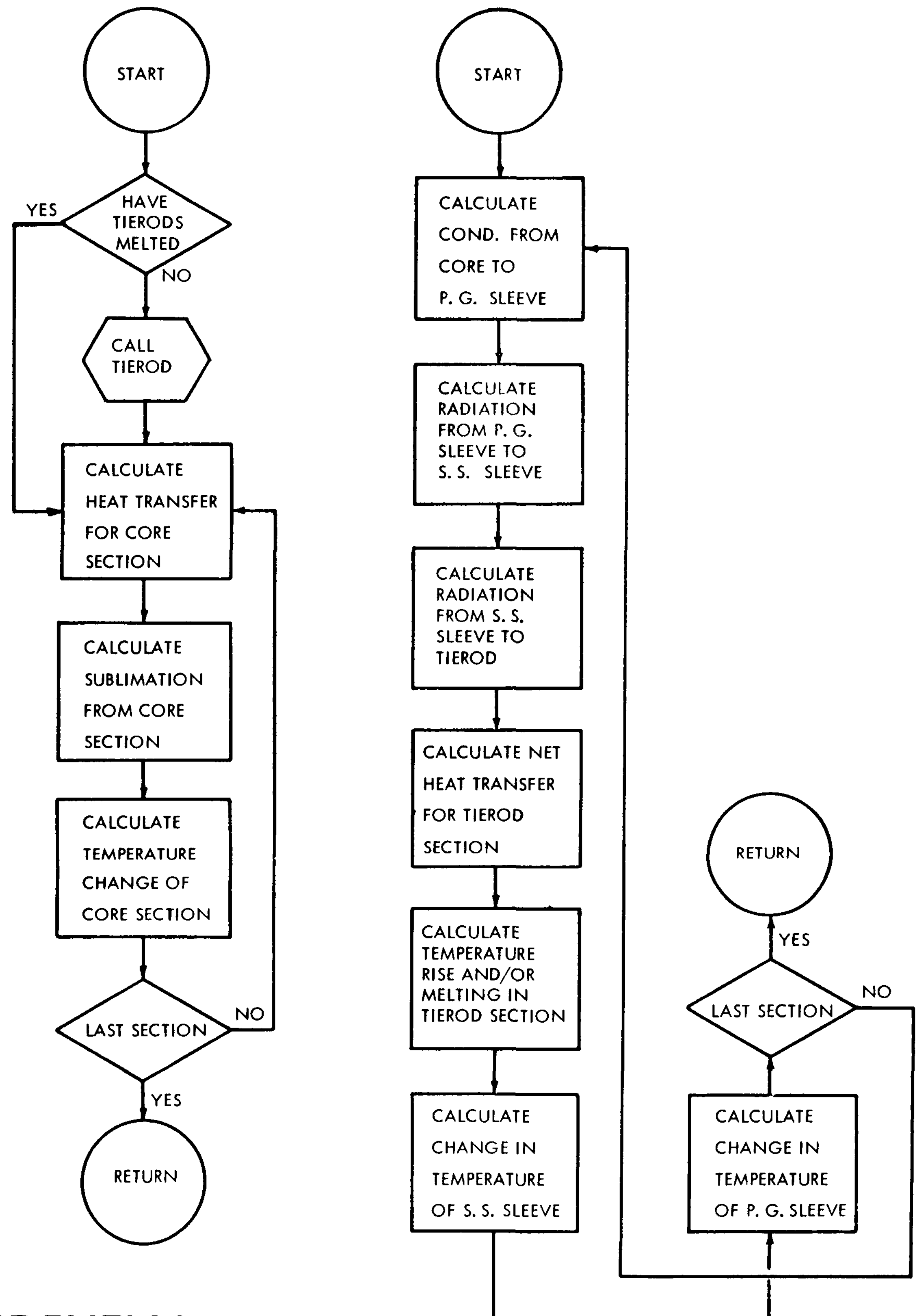
SUBROUTINE N CORE

SUBROUTINE TIEROD



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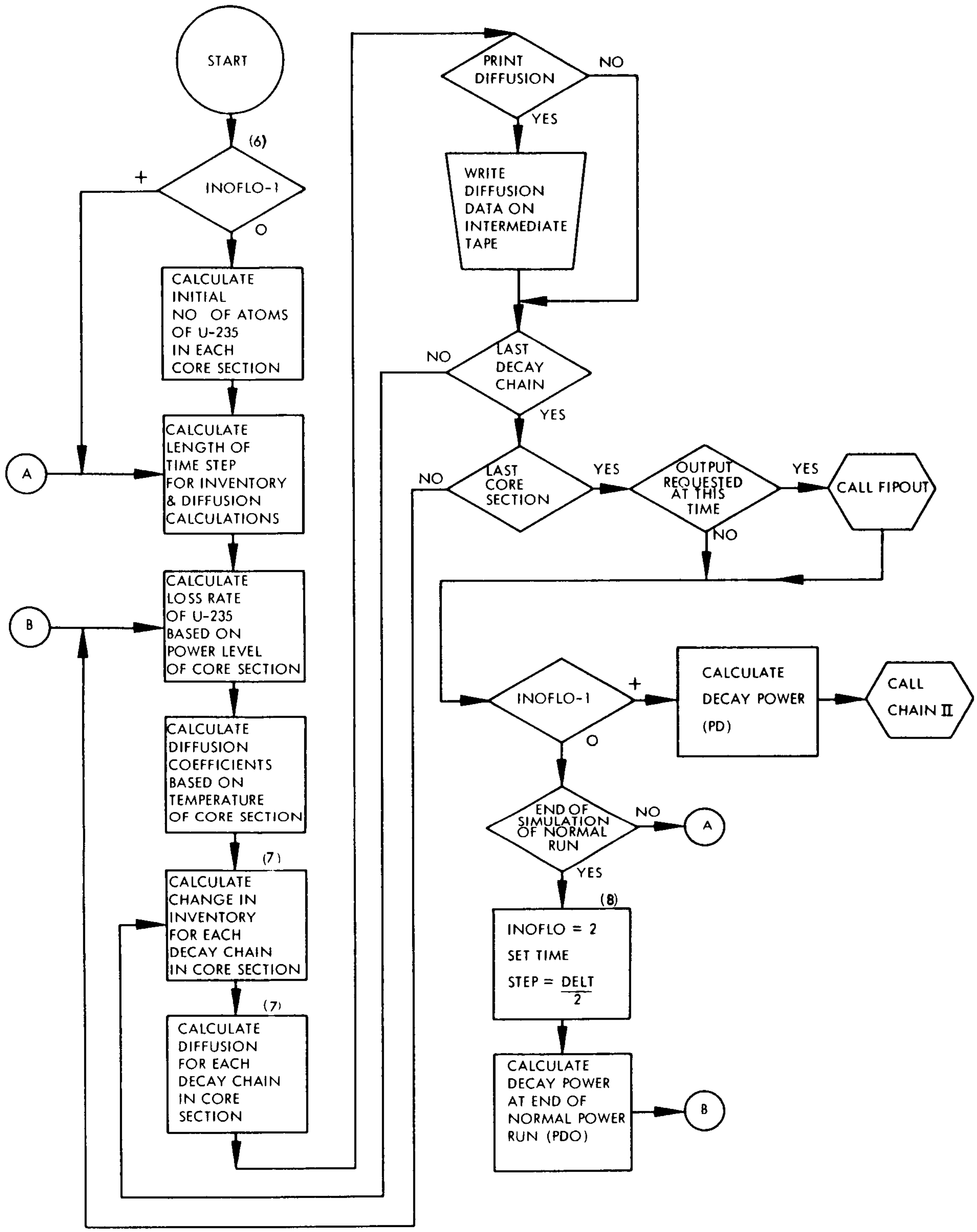
LINK III

LINK III MAIN PROGRAM



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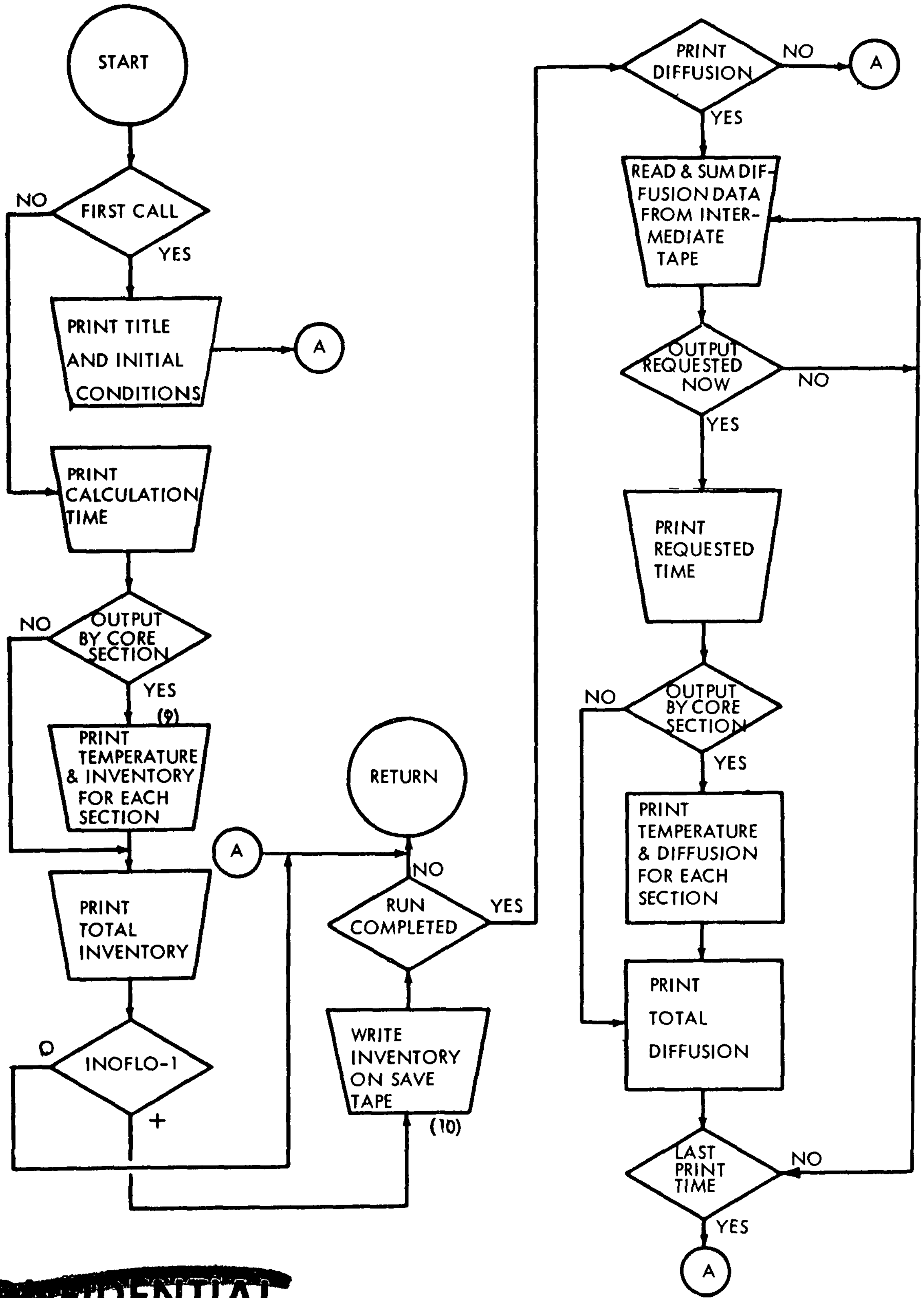
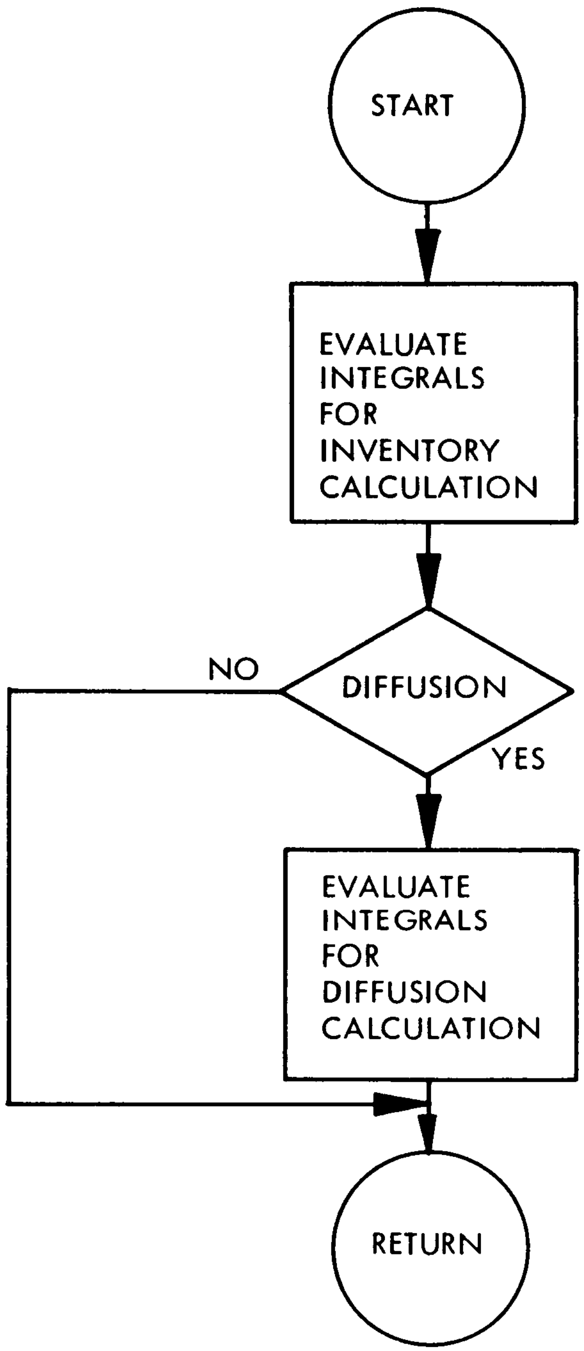
SUBROUTINE FIPOUT



605942-2B

SUBROUTINE EXINT



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VI. NOTES FOR FLOW DIAGRAMS

1. This tape is used as input to a separate program that generates graphs of temperature data via the SC 4020.

2. ALPHA is the exponential decay factor which is calculated from successive decay power determination in LINK III. It is used to extrapolate the decay power for calculation times in LINK II.

3. MTEST is the number of NOFLOW calculations performed between FIPDIF calls.

4. Subroutine CP computes specific heat as a function of temperature and material.

5. Subroutine TTK computer thermal conductivity as a function of temperature and material.

6. INOFLO = 1, normal coolant flow.
   INOFLO = 2, loss of coolant.

7. Integrals for inventory and diffusion calculations are evaluated via subroutine EXINT.

8. DELT = time step to be used in LINK II.

9. Inventory and diffusion are printed for each isotope.

10. This tape is used as input to WANL Source Term Program.

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VII. APPENDIX B

INPUT FOR NOFLOW-FIPDIF

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CARD GROUP	COLUMNS	FORMAT		INPUT QUANTITY
1	1 - 78	13A6		Title Card: Description of Accident to be Simulated
2	1 - 4	I4	INDT	= 1, Read nuclide data from cards
				= 2, Read nuclide data from library tape
	5 - 8	I4	INIC	= 1, No initial fission product inventory exists
				= 2, Read initial fission product inventory from cards
	9 - 12	I4	INDOUT	= 1, Print fission product inventory by section and totals for core
				= 2, Print only total inventory for core
				= 3, Print inventory retained and that released for each section plus core totals
				= 4, Print total inventory retained and that released for core
	13 - 16	I-4	NTCALC	= Number of times a different axial power and temperature distribution are input (≤40)
	17 - 20	I-4	NTOUT	= Number of times output is requested during simulation of normal reactor run (≤40)
	21 - 24	I-4	NTWR	= Number of times output is requested after loss of coolant (≤10)
* 3	1 - 5	I-5	NOCHAN	= Number of decay chains for which inventory calculations are to be made
	6 - 10	I-5	NOGRP	= Number of diffusion groups for "new" fuel
	11 - 15	I-5	NADGR	= Number of diffusion groups for "degraded" fuel
	16 - 25	E10.4	TEMPAD	= Core temperature at which fuel becomes degraded
* 4	1 - 10	E10.0	DO	= Infinite - temperature diffusion constant for diffusion group NG (new fuel)

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CARD GROUP	COLUMNS	FORMAT			INPUT QUANTITY
	11 - 20	E10.0	EQ	=	Ratio of the activation energy, for group NG, to the gas constant (new fuel)
	21 - 30	E10.0	F	=	1.0 except for the two categories used to represent diffusion of isotopes exhibiting a double-rate effect — in this case it is .15 for the category representing diffusion from the pyrocoat and .85 for that representing diffusion from the bead
	Repeat one card for each diffusion group (NOGRP)				
* 5	1 - 10		DOAD	=	Infinite - temperature diffusion constant for diffusion group NG (degraded fuel)
	11 - 20		EQAD	=	Ratio of activation energy, for group NG, to the gas constant (degraded fuel)
	Repeat one card for each diffusion group (NADGR)				
* 6	1 - 80	20I4	NISOT	=	Number of isotopes in the decay chain (including U^{235}, which is considered to be the first member of each decay chain)
	Enter one number per decay chain, use as many consecutive cards as required (NOCHAN)				
* 7	1 - 9	A6, A3	NAME, NAMEND		Alpha - numeric identification of isotope (e.g. Sr-90)
	10 - 19	E10.0	DECON		Decay constant for isotope (sec^{-1})
	20 - 29	E10.0	YIELD		Fractional fission yield
	30 - 39	E10.0	EBETA		Beta decay energy (Mev/disintegration)
	40 - 49	E10.0	EGAM		Gamma decay energy (Mev/disintegration)
	50 - 52	I3	IDI		Diffusion category in which isotope has been placed (new fuel)
	53 - 55	I3	NPAR		Number of isotopes which decay into the given isotope
	56 - 58	I3	NADCAT		Diffusion category (degraded fuel)

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CARD GROUP	COLUMNS	FORMAT		INPUT QUANTITY
* 7	1 - 5	I5	IDPAR(1)	Identification number, within the decay chain, of the first parent of the isotope
	6 - 15	E10.0	B(1)	Branching ratio for formation of the isotope by decay of the first parent
	16 - 20	I5	IDPAR(2)	Identification number of second parent
	21 - 30	E10.0	B(2)	Branching ratio of second parent
	etc.	etc.		etc.

Repeat the two cards in group 7, one for each isotope and each decay chain. If the isotope has no parents, omit the second card.

CARD GROUP	COLUMNS	FORMAT		INPUT QUANTITY
8	1 - 10	E10.0	TCALC(1)	Time at end of first interval during which an axial power and temperature profile is specified
	11 - 20	E10.0	TCALC(2)	Time at end of second interval during which an axial power and temperature profile is specified
	21 - 30	E10.0	TCALC (NTCALC)	NTCALC values should be entered
9	1 - 10	E10.0	TOUT(1)	First time at which output is requested during simulation of normal operation
	11 - 20	E10.0	TOUT(2)	Second time at which output is requested during simulation of normal operation
	21 - 30	E10.0	etc.	NTOUT values should be entered.
10	1 - 10	E10.0	TWRITE(1)	First time at which output is requested during simulation of period following loss of coolant
	11 - 20	E10.0	TWRITE(2)	Second time at which output is requested during simulation of period following loss of coolant
	21 - 30	E10.0	etc.	NTWR values should be entered.

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CARD GROUP	COLUMNS	FORMAT	INPUT QUANTITY	
11	1 - 10	E10.0	P(1,1)	Fission power in first section during first time interval (fissions/sec)
	11 - 20	E10.0	TEMP(1,1)	Temperature (°K) in first section during first time interval
	21 - 30	E10.0	P(1,2)	Fission power in first section during second time interval
	31 - 40	E10.0	TEMP(1,2)	Temperature in first section during second time interval
		etc.		(Ten core sections must be used.)
	Card group 11 is prepared for each core section			
12	1 - 10	E10.0	VSEC(1)	Volume of first core section (cm^3)
	11 - 20	E10.0	WTU25(1)	Weight of U^{235} in first core section (g)
	21 - 30	E10.0	VSEC(2)	Volume of second core section (cm^2)
	31 - 40	E10.0	WTU25(2)	Weight of U^{235} in second core section (g)
		etc.		
13	1 - 4	I4	NDATA1	= 0, Use standard NRX-A temperature distribution corresponding to full power
				= 1, Read temperature distribution for reactor components
	5 - 8	I4	NDATA2	= 0, Use standard power distribution factors
				= 1, Read power distribution factors
	9 - 12	I4	NDATA3	= 0, Use standard geometry constants
				= 1, Read geometry constants
	13 - 16	I4	NDATA4	= 0, Use standard constants for dome and nozzle
				= 1, Read constants for dome and nozzle

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CARD GROUP	COLUMNS	FORMAT	INPUT QUANTITY	
	17 - 20	I4	NPOWER	= 1, Calculate nuclear power only
				= 2, Calculate re-entry heating only
				= 3, Calculate nuclear and re-entry heating
	21 - 25	I4	NSHIELD	= 1, Use Lithium Hydride shield
				= 2, Use Aluminum shield
			METAL	= 1, Use Aluminum pressure vessel and dome
				= 2, Use Titanium pressure vessel and dome
14	1 - 10	E10.4	DELT	Initial time increment in LINK II (sec)
	11 - 20	E10.4	RHO	Shutdown reactivity following coolant loss
	21 - 30	E10.4	TAMB	Atmospheric temperature (°R)
	31 - 40	E10.4	XPRIN	Number of calculations, in LINK II, between printout
	41 - 50	E10.4	DTINC	Maximum allowed change in core temperature during one time step (°R)
	51 - 60	E10.4	DTINC1	Maximum allowed change in component temperatures during one time step (°R)

* The data which is supplied by these cards is presently on a library tape.

[redacted]

VIII. APPENDIX C

NFPLOT is a separate digital computer program that converts the temperature profiles of the core and reactor components into input data for the Stromberg-Carlson SC 4020 plotter. NFPLOT will set up for plotting any or all of the temperature profiles that are generated by the NOFLOW-FIPDIF linked program, as well as a selected number of temperature versus time plots. The axial temperature profiles are plotted on a linear grid and the time versus temperature curves are plotted on a semi-log grid.

The data tape prepared by LINK II of the NOFLOW-FIPDIF Program is used as input to the plotting program. This tape is mounted on tape unit B1 and is 556 CPI.

The NFPLOT Program may be used immediately following the NOFLOW-FIPDIF run. In this case, the intermediate tape prepared by LINK II is left mounted on tape unit B1 following the accident simulation. The data from two or more accident simulations may be placed in sequence on the same plot tape (B6) by setting the indicator LAST = 1.

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INPUT FOR PLOTTING PROGRAM

CARD	COLUMNS	FORMAT		DESCRIPTION
1	1 - 80	13A6,A2	TITLE	Printed on the first line of each plot.
2	1 - 5	I5	NTIME	Number of times to plot axial profiles. If set = 0, program will plot all data on B1.
	10	I1	ICON	= 1 print confidential on all plots.
				= 0 data is unclassified.
	15	I1	LAST	= 1 mount new tape on B1 and continue plotting.
				= 0 end of run.
	20	I1	LIMIT	= 0 semi log grid limits are set by program.
				= 1 left and right semi=log grid limits may be entered.
	21 - 30	E10.4	XLEFT *	left time limit, if set = 0,XLEFT will be set = 10.0 sec.
	31 - 40	E10.4	XRIGHT*	right time limit; if set = 0, XRIGHT will be set = maximum time value on tape B1.
3	1 - 80	8E10.4	TCURVE **	Time after loss of coolant at which axial plots are desired. Enter eight numbers to a card and continue on new cards until NTIME entrys are made.

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* Required only when LIMIT = 1, either the left or right limit may be entered, or both may be entered when limit = 1.

** Card 3 is not required when NTIME = 0.